11th International Conference on Nuclear Criticality safety

Program & Abstracts

September 15-20, 2019

ICNC 2019

Paris, France

Cité des sciences et de l'industrie

Conception : la-fabrique-creative - Photo : © Sylvain Sonnet

HOSTED BY IRSN INSTITUT DE RADIOPROTECTION ET DE SÛRETÉ NUCLÉAIRE

NEA NUCLEAR ENERGY AGENCY

www.icnc2019.com

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Welcome Messages

From IRSN Director General

Jean-Christophe Niel

Twenty years after the 6th ICNC held in France, it is a great honor for IRSN, the French Institute for Radiological Protection and Nuclear Safety, to host the 11th International Conference on Nuclear Criticality safety. This event will be a great opportunity for researchers, experts and practitioners worldwide to share their current works and concerns and also to reflect together about the future challenges of nuclear criticality safety. IRSN is fully mobilized to ensure the success of ICNC 2019.

I express my sincere thanks to my French and international colleagues who already dedicated time and efforts in helping organize this unique and important event, and to all supporting organizations worldwide for making it possible.

From ICNC 2019 Chair

Stéphane EVO

As every edition, the 11th International Conference on Nuclear Criticality safety (ICNC), organized by IRSN under the auspices of the Nuclear Energy Agency of OECD, is a major rendez-vous for professionals and students with activities related to nuclear criticality safety.

Twenty years after, France has the privilege of hosting this conference once again. Even if experience in nuclear criticality safety is important today, the scientific community is still facing new challenges and is improving its skills continuously in order to achieve the highest degree of safety for practitioners dealing with fissile material.

The city of Paris offers a wide variety of activities; we hope you will have the opportunity to extend your stay and make the ICNC 2019 an unforgettable experience.

Organization

LOCAL ORGANIZING COMMITTEE

GENERAL CHAIR
Stéphane EVO, *IRSN*

ASSISTANT GENERAL CHAIRS
Aurélie BARDELAY, *IRSN*
Shuichi TSUDA, *OECD/NEA*

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Fabrice ECRABET, *IRSN*
Caroline SALHAB, *IRSN*

PCO
Audrey DUPUIS, *Insight Outside*

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Sylvie GRAUET, *IRSN*
Caroline MARCHAND, *IRSN*
Magali PENOT, *IRSN*

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José CONDE, *ENUSA*
Jim GULLIFORD, *G&M Nuclear Skills Ltd*
Axel HOEFER, *Framatome*
Deborah HILL, *NNL*
Tatiana IVANOVA, *OECD/NEA*
Anatoly KOCHETKOV, *SCK CEN*
Dennis MENNERDAHL, *EMS*
Ken NAKAJIMA, *RRI*
Gregory O'CONNOR, *ONR*
Cecil PARKS, *ORNL*
Catherine PERCHER, *LLNL*
Anssu RANTA-AHO, *TVO*
Brad REARDEN, *ORNL*
Boris RYAZANOV, *IPPE*
Maik STUKE, *GRS*
Kotaro TONOIKE, *JAEA*

INTERNATIONAL TECHNICAL PROGRAM COMMITTEE (ITC)

Stéphane EVO, Chair of the Organizing Committee, *IRSN, France,*

TRACK LEADERS

Track 1. Codes and Other Calculation Methods

International Track Leaders
David HEINRICHS, *LLNL, United States*
Yevgeniy ROZHIKHIN, *IPPE, Russia*
French Track Leader
Eric DUMONTEIL, *IRSN, France*

Track 2. Nuclear Data

International Track Leaders
David BROWN, *BNL, United States*
Luiz LEAL, *IRSN, France*
French Track Leader
Raphaëlle ICHOU, *IRSN, France*

Track 3. Uncertainty and Sensitivity Analysis

International Track Leaders
Coralie CARMOUZE, *CEA, France*
Maik STUKE, *GRS, Germany*
French Track Leader
Nicolas LECLAIRE, *IRSN, France*

Track 4. Measurements, Experiments and Benchmarks

International Track Leaders
John BESS, *INL, United States*
Patrick BLAISE, *CEA, France*
French Track Leader
Isabelle DUHAMEL, *IRSN, France*

Track 5. Standards, Assessment Methodology, Regulations

International Track Leaders
Doug BOWEN, *ORNL, United States*
Fred WINSTANLEY, *SL, United Kingdom*
French Track Leader
Luis AGUIAR, *IRSN, France*

Track 6. Operational Practices and Safety Cases

International Track Leaders
Boris RYAZANOV, *IPPE, Russia*
Stuart WATSON, *3TSC, United Kingdom*
French Track Leader
Mathieu MILIN, *IRSN, France*

Track 7. Storage and Transport Issues

International Track Leaders

Gregory O'CONNOR, *ONR, United Kingdom*
Marcel TARDY, *ORANO TN, France*

French Track Leader

Ludyvine JUTIER, *IRSN, France*

Track 8. Final Disposal Issues

International Track Leaders

Travis TATE, *US NRC, United States*
Vladimir KHOTYLEV, *CNSC, Canada*

French Track Leader

Grégory CAPLIN, *ORANO Projects, France*

Track 9. Criticality Accidents and Incidents

International Track Leaders

Yuichi YAMANE, *JAEA, Japan*
Matthieu DULUC, *IRSN, France*

Track 10. Professional Development Issues and Training

International Track Leaders

Deborah HILL, *NNL, United Kingdom*
John MILLER, *SNL, United States*

French Track Leader

Céline LENEPVEU, *IRSN, France*

Track 11. Future Challenges

International Track Leaders

Angela CHAMBERS, *NNSA, United States*
Tatiana IVANOVA, *NEA, France*

French Track Leader

Jean-Baptiste CLAVEL, *IRSN, France*

REVIEWERS

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Marcel TARDY, *Orano, France*
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Anthony WILSON, *SL, United Kingdom*
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Yuichi YAMANE, *JAEA, Japan*
Andrea ZOIA, *CEA, France*
Oscar ZURRON, *ENUSA, Spain*
Will ZYWIEC, *LLNL, United States*

PROFESSIONAL CONGRESS ORGANIZER (PCO)

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www.insight-outside.fr • Twitter: @_InsightOutside

Supporter Information

Studsvik

340 Tschiffely Square Rd, Gaithersburg, MD 20878 USA
www.studsvik.com/nfa

Studsvik is a well-established, highly regarded company that has been active in the nuclear field for more than 60 years. Studsvik Scandpower (SSP) is the industry leading provider of software tools in the field of reactor physics, reactivity management, and spent nuclear fuel management for Light Water Reactors (LWRs). Studsvik's methods are used in all the countries where nuclear energy is a contributor to electricity production. They have been used to model more than 200 LWRs contributing to analyzing far more than 1000 reactor cycles for both PWR and BWR reactors. Regulators, utilities, fuel vendors, research institutes, as well as new reactor designers are users of Studsvik's codes. CASMO, SIMULATE, S3K, S3R, SNF, MARLA and GARDEL are the benchmark by which others measure the accuracy of their methods. SSP is dedicated to the development, maintenance and support of its products to ensure efficient, accurate, and easy to use software. SSP's business focus on needs of the nuclear industry requires that we continue to invest in our products to guarantee they continue to be the reference in the industry.

Commissariat à l'Energie Atomique et aux Energies Alternatives

CEA siège, 91191 Gif-sur-Yvette Cedex
www.cea.fr

The French Alternative Energies and Atomic Energy Commission (CEA) is a key player in research, development and innovation in four main areas:

- Defence and security,
- Low carbon energies (nuclear and renewable energies),
- Technological research for industry,
- Fundamental research in the physical sciences and life sciences.

The CEA is established in nine centers spread throughout France. It works in partnership with many other research bodies, local authorities and universities. Within this context, the CEA is a stakeholder in a series of national alliances set up to coordinate French research in energy (ANCRE), life sciences and health (AVIESAN), digital science and technology (ALLISTENE), environmental sciences (AllEnvi) and human and social sciences (ATHENA).

Widely acknowledged as an expert in its areas of skill, the CEA is actively involved in the European Research Area and its international presence is constantly growing.

ENSTTI

12 rue de la Redoute 92260 Fontenay-aux-Roses, France
www.enstti.eu

ENSTTI is a professional training and tutoring institute. Its mission is to share the knowledge and expertise of the European nuclear safety organizations. The ENSTTI initiative was set up in 2010 to meet the growing need for trained experts, prompting the major European technical safety organizations for the nuclear industry (which are also members of the ETSON network) to pool their resources. Every year, more than a thousand participants enter its training and tutoring program.

Partners: The European Commission through its Instrument for Nuclear Safety Cooperation Training and Tutoring (INSC T&T), the IAEA Technical Cooperation Program, and the IAEA Nuclear Safety and Security Department (among other programs and resources), rely on ENSTTI to provide their beneficiary organizations and countries with training and tutoring in nuclear safety/security and radiation protection.

Catalogue :

- The 2019 curriculum comprises 37 courses;
- These are organized in 25 sessions that take place in Europe and elsewhere.

BWX Technologies

P. O. Box 785, Lynchburg, VA 24505, USA
www.bwxt.com

BWX Technologies, Inc. (BWXT) is a leading supplier of nuclear components and fuel to the U.S. government; provides technical and management services to support the U.S. government in the operation of complex facilities and environmental remediation activities; and supplies precision manufactured components, services and fuel for the commercial nuclear power industry. With approximately 6,350 employees, BWXT has 11 major operating sites in the U.S. and Canada. In addition, BWXT joint ventures provide management and operations at more than a dozen U.S. Department of Energy and NASA facilities. Learn more at www.bwxt.com.

Exhibitors Information

Commissariat à l'Energie Atomique et aux Energies Alternatives — Nuclear Energy Division

CEA siège, 91191 Gif-sur-Yvette Cedex, France

www.cea.fr

Within the CEA, the Nuclear Energy Division (DEN) provides the French government and industry with technical expertise and innovation in nuclear power generation systems to develop sustainable nuclear energy that is both safe and economically competitive.

To meet these objectives, the DEN is engaged in three main areas of investigation:

- Optimising the current nuclear industry;
- Developing nuclear systems of the future – dubbed "4th generation" reactors – and their fuel cycles;
- Developing and operating large experimentation and simulation tools needed for its research programmes.

As nuclear operator, the DEN also has to manage and upgrade its own fleet of nuclear facilities. It carries out numerous construction and refurbishment programmes on its facilities, together with clean-up and dismantling programmes for those that have reached the end of their service life.

Cristal V2

www.cristal-package.org

CRISTAL V2 includes 4 criticality calculation routes allowing multigroup and continuous energy calculations: two multigroup routes based on multigroup (281 groups) cross-sections (APOLLO2 – MORET 5 or APOLLO2 Sn calculations), a point-wise Monte Carlo route (TRIPOLI-4®) and a criticality standard calculation route.

CRISTAL V2 package includes all the elements necessary for criticality calculations for nuclear fuel cycle facilities (fabrication, reprocessing, etc.), storage and transportation of fissile materials. These elements include:

- The basic nuclear data, including microscopic cross sections.
- Computer codes (APOLLO2, TRIPOLI-4®, MORET 5).
- Graphical User Interface (LATEC).
- Recommended calculation options and calculation procedures.

International Organization for Standardization (ISO) – Working Group 8

TC85/SC5 secretary, Sellafield Ltd., Seascale, Cumbria, CA20 1PG, United Kingdom

www.iso.org/committee/50328.html

Within the Sub-Committee 5 (SC5, « Nuclear installations, processes and technologies ») of the Technical Committee 85 (TC85, « Nuclear energy, nuclear technologies, and radiological protection ») of the International Organization for Standardization (ISO), the Working Group 8 (WG8, "Nuclear criticality safety") is in charge of developing and maintaining international standards relative to the prevention of nuclear criticality accidents and to the limitation of the consequences of such accidents should they occur. WG8 is led by M. Doug BOWEN and is composed of 34 members representing 11 countries. There are currently 8 published standards and several projects under development (among which 3 are at the final stages of development).

Institut de Radioprotection et de Sûreté Nucléaire (IRSN)

31, avenue de la Division Leclerc, 92260 Fontenay-aux-Roses, France

http://dosimetrie.irsn.fr/fr-fr

IRSN dosimetry lab has been the French reference for more than 50 years for occupational dosimetry service.

170 000 monitored customers - 1 250 000 analyzes per year. IRSN Dosimetry lab is a high capacity and high technology facility, able to offer a complete service for customers with a complete range of dosimeters: whole body – extremity – eye lens dosimeters, individual and environmental.

With Radio Photo Luminescence (RPL) Technology IRSN is the leader in France; it also provides TLD dosimeters for eye lens and extremities.

For the Criticality dose measurement, IRSN dosimetry lab offers a complete system with a criticality belt, an individual dosimeter (RPL neutron criticality) and a specific area neutron spectrometer.

SDEC France

Z.I de la Gare, CS 50027 – Tauxigny, 37310 Reignac-sur-Indre, France

www.sdec-france.com

Since 1991, SDEC France has been a specialist in measurement instrumentation for the environment: air - water - soil.

As a leader in Tritium and Carbon-14 sampling, SDEC France offers a complete range of solutions for radioprotection. Our services: Study, custom design, industrialization, product consulting, training, on-site maintenance.

They trust us: IAEA, CEA, EDF, IRSN, AREVA, ANDRA, MARINE NATIONALE...

Silver Fir Software

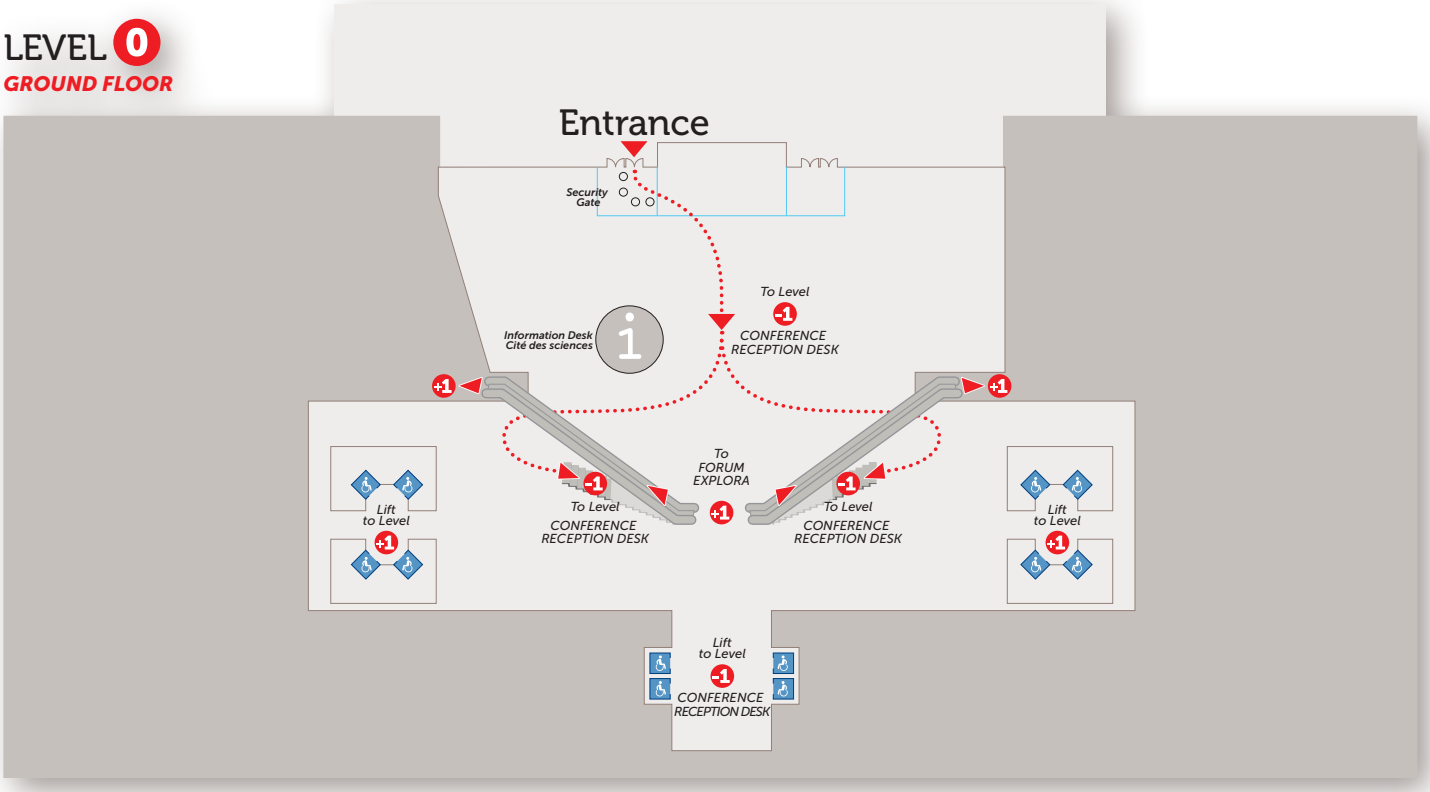
6659 Kimball Drive Ste E502, Gig Harbor, WA, 98335, United States

http://www.silverfirsoftware.com

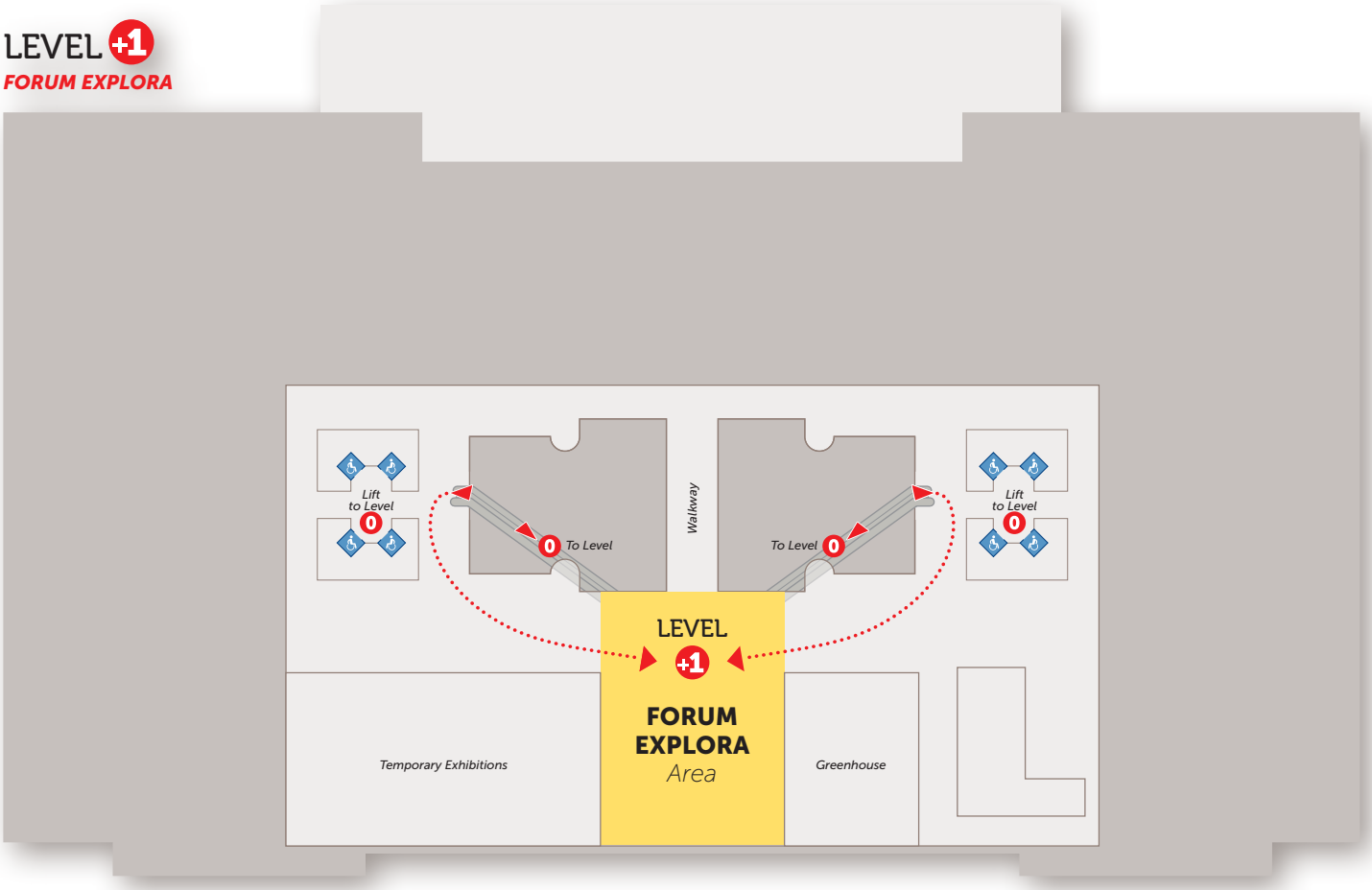
Silver Fir Software is an independent company focusing exclusively on improving user productivity for radiation transport simulations. Monte Carlo and deterministic radiation transport solvers are mostly developed by government agencies, with a heavy emphasis on numerical methods and little attention to improving the user experience. Our principal goal is to help organizations get the most out of their radiation transport simulations by developing user friendly software to shorten and simplify the analysis cycle. Our Attila4MC product provides users with a graphical user interface front end for MCNP® that can eliminate the most time-consuming bottlenecks in setting up, running, and visualizing Monte Carlo

Meeting Spaces

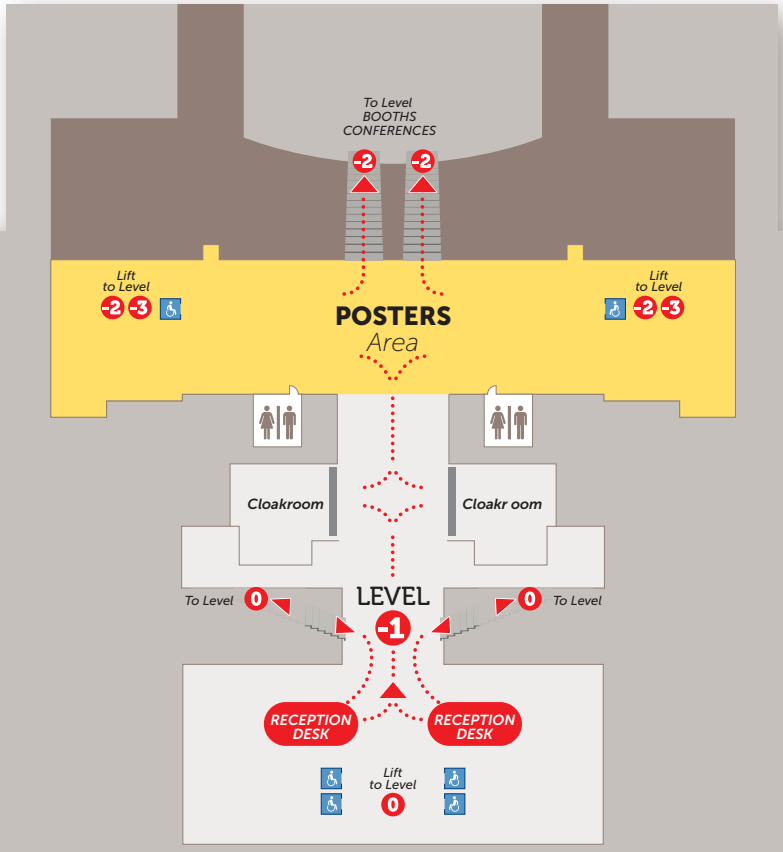
LEVEL 0
GROUND FLOOR



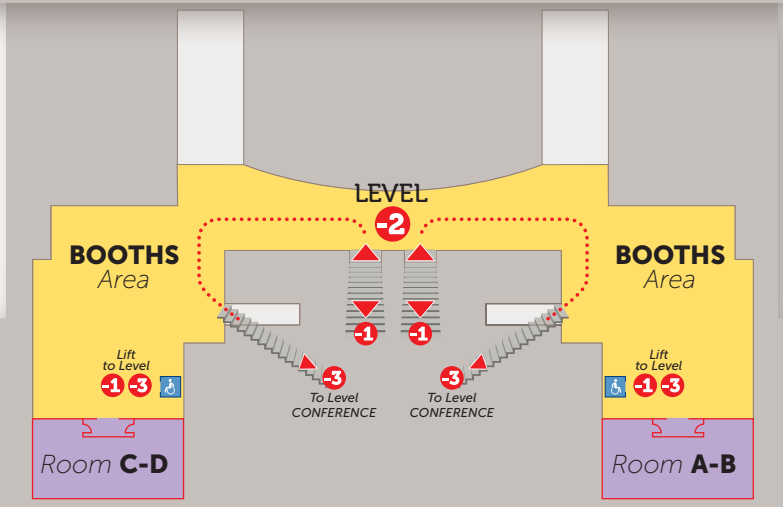
LEVEL +1
FORUM EXPLORA



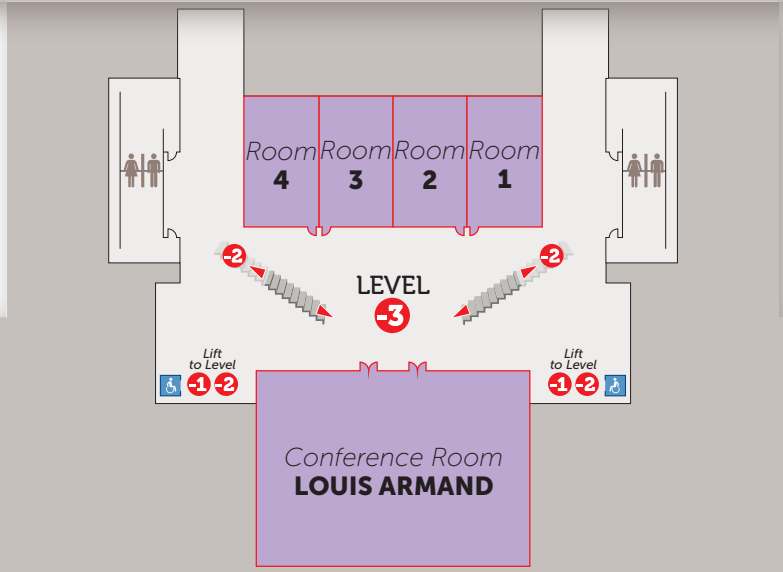
LEVEL -1
RECEPTION / POSTERS Area



LEVEL -2
BOOTHS / CONFERENCE Area



LEVEL -3
CONFERENCE Area



Program at a Glance

SUNDAY, SEPTEMBER 15

15h00/20h00	Registration -1 RECEPTION DESK
18h00/20h00	Welcome Reception -1 POSTERS Area

MONDAY, SEPTEMBER 16

8h30/9h00	Welcome Coffee -2 BOOTHS / CONFERENCE Area
9h00/11h30	Opening Ceremony -3 Conference Room LOUIS ARMAND

Coffee Break

	Session 1 -2 Room A - B	Session 2 -2 Room C - D	Session 3 -3 Room 1	Session 4 -3 Conference Room LOUIS ARMAND
12h00/12h50	**Track 6** Operational Practices and Safety Cases	**Track 2** Nuclear Data	**Track 1** Codes and Other Calculation Methods	**Track 7** Storage and Transport Issues
Lunch				
14h00/15h40	**Track 6** Operational Practices and Safety Cases	**Track 2** Nuclear Data	**Track 1** Codes and Other Calculation Methods	**Track 7** Storage and Transport Issues
Coffee Break				
16h10/17h50	**Track 6** Operational Practices and Safety Cases	**Track 2** Nuclear Data	**Track 1** Codes and Other Calculation Methods	**Track 4** Measurements, Experiments and Benchmarks

TUESDAY, SEPTEMBER 17

8h30/9h00	Welcome Coffee -2 BOOTHS / CONFERENCE Area

	Session 1 -2 Room A - B	Session 2 -2 Room C - D	Session 3 -3 Room 1	Session 4 -3 Conference Room LOUIS ARMAND
9h00/10h40	**Track 6** Operational Practices and Safety Cases	**Track 3** Uncertainty and Sensitivity Analysis	**Track 1** Codes and Other Calculation Methods	**Track 7** Storage and Transport Issues
Coffee Break				
11h10/12h50	**Track 6** Operational Practices and Safety Cases	**Track 3** Uncertainty and Sensitivity Analysis	**Track 4** Measurements, Experiments and Benchmarks	**Track 7** Storage and Transport Issues
Lunch				
14h00/16h05	**Track 5** Standards, Assessment Methodology, Regulations	**Track 2** Nuclear Data	**Track 9** Criticality Accidents and Incidents	**Track 8** Final Disposal Issues
Coffee Break				
16h10/17h50	Poster Session -1 POSTERS Area			

WEDNESDAY, SEPTEMBER 18

8h30/9h00 — Welcome Coffee
-2 BOOTHS / CONFERENCE Area

	Session 1 -2 Room A - B	Session 2 -2 Room C - D	Session 3 -3 Room 1	Session 4 -3 Conference Room LOUIS ARMAND
9h00/10h40	**Track 5** Standards, Assessment Methodology, Regulations	**Track 3** Uncertainty and Sensitivity Analysis	**Track 1** Codes and Other Calculation Methods	**Track 4** Measurements, Experiments and Benchmarks
	Coffee Break			
11h10/12h50	**Track 5** Standards, Assessment Methodology, Regulations	**Track 3** Uncertainty and Sensitivity Analysis	**Track 9** Criticality Accidents and Incidents	**Track 4** Measurements, Experiments and Benchmarks
	Lunch			
14h00/15h40	**Track 5** Standards, Assessment Methodology, Regulations	**Track 10** Professional Development Issues and Training	**Track 9** Criticality Accidents and Incidents	**Track 4** Measurements, Experiments and Benchmarks
	Coffee Break			
16h10/17h50	**Track 11** Future Challenges	**Track 10** Professional Development Issues and Training	**Track 9** Criticality Accidents and Incidents	**Track 4** Measurements, Experiments and Benchmarks

19h00/20h00 — Cocktail
+1 FORUM EXPLORA

20h00/23h00 — Gala Dinner
+1 FORUM EXPLORA

THURSDAY, SEPTEMBER 19

8h30/9h00 — Welcome Coffee
-2 BOOTHS / CONFERENCE Area

	Session 1 -2 Room A - B	Session 2 -2 Room C - D	Session 3 -3 Room 1	Session 4 -3 Conference Room LOUIS ARMAND
9h00/10h40	**Track 11** Future Challenges	**Track 10** Professional Development Issues and Training	**Track 9** Criticality Accidents and Incidents	**Track 4** Measurements, Experiments and Benchmarks
	Coffee Break			

11h10/12h40 — Closing Ceremony
-3 Conference Room LOUIS ARMAND

12h40/13h00 — Departure for the Technical Tours
-1 RECEPTION DESK

Workshops

	-3 Room 3	-3 Room 4
14h00/15h20	**Workshop 1** Second Level Criticality Modelling with CRISTAL Package: Enhancing Criticality Safety Assessments for Industrial Applications	**Workshop 2** Enhancing Validation of Nuclear Criticality safety Calculations with ICSBEP Handbook an NEA Tools
	Coffee Break	
15h40/17h00	**Workshop 1** Second Level Criticality Modelling with CRISTAL Package: Enhancing Criticality Safety Assessments for Industrial Applications	**Workshop 2** Enhancing Validation of Nuclear Criticality safety Calculations with ICSBEP Handbook an NEA Tools

Program Schedule

SUNDAY, SEPTEMBER 15

15h00/20h00
Registration
-1 ***RECEPTION DESK***

18h00/20h00
Welcome Reception
-1 ***POSTERS*** *Area*

MONDAY, SEPTEMBER 16

9h00/11h30
Opening Ceremony
-3 *Conference Room* ***LOUIS ARMAND***

Coffee Break

12h00/12h50

Session 1 -2 *Room* ***A - B***	Session 2 -2 *Room* ***C - D***	Session 3 -3 *Room* ***1***	Session 4 -3 *Conference Room* ***LOUIS ARMAND***
Track 6 OPERATIONAL PRACTICES AND SAFETY CASES	**Track 2** NUCLEAR DATA	**Track 1** CODES AND OTHER CALCULATION METHODS	**Track 7** STORAGE AND TRANSPORT ISSUES
The Disposal of the Final Concentrate Mother Liquor from THORP: How Criticality Safety Analysis can Impact Overall Risk *A. SUTTON*	Development and Implementation of an Improved Liquid TSL Treatment in the FLASSH Code *C.A. MANRING, A.I. HAWARI*	Solomon: a Monte Carlo Solver for Criticality Safety Analysis *Y. NAGAYA, T. UEKI, K. TONOIKE*	Evaluation of the Impact of Neutron Absorber Material Blistering and Pitting on Spent Fuel Pool Reactivity *H. AKKURT, M.WENNER, A. BLANCO*
Recent Improvements in the K-Area Criticality Safety Program *B. WILLIAMSON, J.S. BAKER*	Measurement of the Double-Differential Neutron Cross Section of Uo2 from Room Temperature to Hott Full Power Conditions *S. XU, G. NOGUERE, A. FILHOL et al.*	Development of Terrenus, a Multiphysics Code for Spent Nuclear Fuel Cask Criticality Analysis *G.G. DAVIDSON, S.R. JOHNSON, S. CHATZIDAKIS et al.*	Some Insights in Criticality-Safety of Spent Fuel Pools under Loss-of-Cooling and Loss-of-Coolant Accident *L. JUTIER, T. ALBERT, O. DE LUZE*

Lunch

MONDAY, SEPTEMBER 16

14h00 / 15h40

Session 1	Session 2	Session 3	Session 4
-2 Room A - B	-2 Room C - D	-3 Room 1	-3 Conference Room LOUIS ARMAND
Track 6 OPERATIONAL PRACTICES AND SAFETY CASES	**Track 2** NUCLEAR DATA	**Track 1** CODES AND OTHER CALCULATION METHODS	**Track 7** STORAGE AND TRANSPORT ISSUES
The Benefit of Different Approaches to Bounding Poison Quantification *S.A. WATSON*	International Benchmarks Intercomparison Study for Code and Nuclear Data Validation *I. DUHAMEL, J.L. ALWIN, F.B. BROWN et al.*	SIMULATE5 Analysis of a Spent Fuel Pool *J. HYKES, T. BAHADIR, D. DEAN et al.*	Criticality Analysis of the New DN30 Package for the Transport of UF_6 *M. HENNEBACH, F. HILBERT*
Cleaning and Dismantling of Hot Cells Dedicated to Mecanichal Treatment and Shearing of Spent Fuel – Criticality Safety Issues *L. CHOLVY, C. FABRY*	Evaluation Updates for Major and Minor Actinides *I. STETCU, T. KAWANO, D. NEUDECKER et al.*	TRIPOLI-4® : Overview of the Code Capabilities for Criticality-Safety in Version 11 *E. BRUN, F-X. HUGOT A. JINAPHANH et al.*	Assessing the Effects of Low Temperatures on K-effective for AGR Spent Fuel Transport Packages *J.D. WATSON, J.S. MARTIN, M. HENDERSON et al.*
Homogenization Techniques for Bounding Criticality Safety Analyses for Fuel Fabrication and Repair *A. HOEFER, O. BUSS, S. GLAUBRECHT et al.*	Testing New Thermal Scattering Law for Light Water at 600 K using VESTA 2.2 Depletion Calculations *R. ICHOU, V. JAISWAL, L. LEAL et al.*	Recent Developments to the MONK Monte Carlo Code for Criticality Safety and Reactor Physics Analyses *S. RICHARDS, G. DOBSON, T. FRY et al.*	Effect of Low Temperatures on Criticality Calculation for the Transport of Fissile Material *M. MILIN, C. POULLELAOUEN, R. ICHOU et al.*
A Simple Alternative Approach for the Modelling of Fuel Assemblies with Missing Fuel Rods *T. ALBERT, A. BARDELAY, L. AGUIAR et al.*	Progress on the RECONR module for NJOY21 *W. HAECK, A.P MCCARTNEY, J.L. CONLIN et al.*	Evaluation of MCNP's Fission Matrix Capability for Criticality Calculations *S. HENDERSON, J.A. MILLER, F. BROWN*	AWG-711, a type C transport package *W. PHILPOTT, R. JONES*

Coffee Break

16h10 / 17h50

Track 6 OPERATIONAL PRACTICES AND SAFETY CASES	**Track 2** NUCLEAR DATA	**Track 1** CODES AND OTHER CALCULATION METHODS	**Track 4** MEASUREMENTS, EXPERIMENTS AND BENCHMARKS
Integration of Uncertainties into the Safety Analysis for a Large Number of Movements *O. RAVAT*	Measurement of Gamma Rays from Radiative Capture of Uranium-238 and Decay of Short Lived Fission Products from Subcritical System *Y. NAUCHI, T. SANO, H. UNESAKI et al.*	Automated Acceleration and Convergence Testing for Monte Carlo Nuclear Criticality Safety Calculations *F. BROWN, C. JOSEY, S. HENDERSON et al.*	Investigation of Inferred Parameters in Subcritical Experiments *J. HUTCHINSON, J. ARTHUR, R. BAHRAN et al.*
Nuclear Criticality Safety Assessment Supporting the Integrated Safety Analysis of the Pellet Fabrication Process at the Juzbado Plant *J. LOPEZ-MARQUEZ, C. PAREDES-HAYA, O. ZURRÓN-CIFUENTES*	Impact of Experimental Correlation on Transposition Method Carry out with Critical Integral Experiments *T. NICOL, C. CARMOUZE*	Critical Experiment Design using Optimus *J. NORRIS*	Measurements of Subcriticality in Dollar Units using Time-Domain Decomposition Based Integral Method *A. NONAKA, T. ENDO, A. YAMAMOTO et al.*
Uranium Accumulations in Casting Operations *T.L WILSON, S.P. JORDAN*		Validation of Deep Learning Methods for Nuclear Criticality Safety *W. ZYWIEC, A.J. NELSON*	MUSiC: A Critical and Subcritical Experiment Measuring Highly Enriched Uranium Shells *A. MCSPADEN, T. CULTER, J. HUTCHINSON et al.*
Managing the Risk from Flooding for a Facility at AWE *E. WATSON, M. A. ROYDHOUSE, J. VENNER*		Neutron Multiplication in Fuel-Water Random Media *P. BOULARD, C. LARMIER, J.C. JABOULAY et al.*	

End of Sessions

TUESDAY, SEPTEMBER 17

9h00/10h40

Session 1 (-2 Room A - B)	Session 2 (-2 Room C - D)	Session 3 (-3 Room 1)	Session 4 (-3 Conference Room LOUIS ARMAND)
Track 6 OPERATIONAL PRACTICES AND SAFETY CASES	**Track 3** UNCERTAINTY AND SENSITIVITY ANALYSIS	**Track 1** CODES AND OTHER CALCULATION METHODS	**Track 7** STORAGE AND TRANSPORT ISSUES
The Double Control and its Consistency with the Double Contingency Principle *G. KYRIAZIDIS, P. RIEPPERT*	Effect and Uncertainties of H in Ice Thermal Scattering Laws on the Neutron Multiplication Factor for PWR Fuel Criticality Applications *M. TIPHINE, C. CARMOUZE, G. NOGUERES, F. CANTARGI et al.*	Validation of the Burn-up Code MOTIVE Using ENDF/B-VIII Data *V. HANNSTEIN, M. BEHLER, F. SOMMER*	A Misload Analysis Methodology Supporting Criticality Analysis of Spent Nuclear Fuel Canisters Using As-Loaded Configurations *K. BANERJEE, H. LILJENFELDT, J.B. CLARITY et al.*
Development of a UK Working Party on Criticality Learning from Experience Database *M. ERLUND, A. BROWN, M. SAVAGE et al.*	Representativity Analysis in Reactor Core Calculations *P. LOPEZ, A. BIDAUD, D. PORTINARI*	Interpretation of GEDEON-1 and GEDEON-2 Gadolinium Depletion Experimental Analysis with the DARWIN2.3 Package *T. NICOL, D. BERNARD*	Criticality Safety Analysis of Spent Nuclear Fuel Canisters Using As-Loaded Configurations *K. BANERJEE, J.B. CLARITY, H. LILJENFELDT et al.*
Nuclear Criticality Safety Lessons Learned in the Design of the Uranium Processing Facility at the Y-12 National Security Complex *K. REYNOLDS*	Sensitivity and Uncertainty Based Techniques to Extend the Database of Experimental Validation Benchmarks: Practical Example of Use for TRIGA® Fuel *C. RECHATIN, Q. VUYET, N. COMTE et al.*	Verification and Validation of the Depletion Capability of the High-Fidelity Neutronics Code NECP-X *X. WEN, Z. LIU, K. HUANG et al.*	On the Benefits to Take Account of the Depletion of Fast-Neutron Reactor Fuel Elements for Transportation *C. CARMOUZE, M. TARDY, G. GRASSI et al.*
	Whisper S/U Benchmark Analysis of Metal-Water Critical Mass Curves *W. COOK, J.A. MILLER, S. HENDERSON et al.*	High Fidelity KCODE Modeling of Subcritical Benchmarks Using MCNP 6.2 *D. TIMMONS, M. RISING, C. PERFETTI*	Criticality Safety Analysis for Storage and Transportation Applications Using NRC ISG-8 Rev. 3 *R. MIGLIORE, J. LI, P.T.T. PHAM*

Coffee Break

11h10/12h50

Track 6 OPERATIONAL PRACTICES AND SAFETY CASES	Track 3 UNCERTAINTY AND SENSITIVITY ANALYSIS	Track 4 MEASUREMENTS, EXPERIMENTS AND BENCHMARKS	Track 7 STORAGE AND TRANSPORT ISSUES
The Use of a Hand-Held Enrichment Device in Support of Uranium Residue Recovery - a Benefit or False Confidence ? *D. HILL*	Use of Whisper S/U Techniques in Support of Benchmark Identification *J.A. MILLER, W.M. COOK, S. HENDERSON et al.*	Fundamental Physics Subcritical Neutron Multiplicity Benchmark Experiments Using Water Moderated Highly Enriched Uranium Fuel *A.J. NELSON, W. MONANGE, S.S. KIM et al.*	Determination of Bounding Axial Burnup Distributions for PWR Spent Fuel Assemblies Discharged from Nuclear Power Plants in South Korea *K.J. CHOI , D.J. KIM, Y.S. CHO et al.*
Nuclear Criticality Safety Analysis: Recovery of Old Containers Holding Fissile Material *E. FILLASTRE, A. DORVAL, L. MANDARD et al.*	Parametric Analysis of Handbook Metal-Water Critical Mass Curves with MCNP *W.M. COOK, J.A. MILLER, S. HENDERSON et al.*	Sub-Criticality Monitoring System for the Retrieval of Fuel Debris in Fukushima Dai-ichi Nuclear Power Plants *S. WADA, S. KANO, T. MISAWA, et al.*	Overview of the Recent BWR Burnup Credit Project at Oak Ridge National Laboratory *W. MARSHALL, B.J. ADE, I.C. GAULD et al.*
Improved Safety Basis for Liquid Waste Processing at BWXT *L. WETZEL*	Tools for Validation and Uncertainty Quantification with ANSWERS Software *P. SMITH , D. HANLON, G. DOBSON et al.*	Validation of MCNP® Rossi-Alpha Calculations Using Recent Measurements *G. MCKENZIE*	Burnup Credit Implementation for Enriched Reprocessed Uranium Used Fuel Transportation *L. MILET, M. TARDY, D. LIN et al.*
Development of a Criticality Safety Case for Waste Retrieval from a Historical Waste Storage Facility *M. HOBSON*	Evaluating Sensitivity-based Similarity Metrics between Applications and Benchmarks *M. RISING*	Conversion from Prompt Neutron Decay Constant to Subcriticality Using Point Kinetics Parameters Based on Alpha- and k-eigenfunctions *T. ENDO, A. YAMAMOTO*	Using the ORNL Spent Fuel Database Tool UNF-STAD&RDS for as Loaded and Scooping Calculations for the Swedish Spent Nuclear Fuel Repository *F. JOHANSSON, H. LILJENFELDT*

Lunch

TUESDAY, SEPTEMBER 17

14h00 / 16h05

Session 1	Session 2	Session 3	Session 4
-2 Room **A - B**	-2 Room **C - D**	-3 Room **1**	-3 Conference Room **LOUIS ARMAND**
Track 5 STANDARDS, ASSESSMENT METHODOLOGY, REGULATIONS	**Track 2** NUCLEAR DATA	**Track 9** CRITICALITY ACCIDENTS AND INCIDENTS	**Track 8** FINAL DISPOSAL ISSUES
Periodic Safety Review in France - Focus on Nuclear Criticality Safety *M. DULUC, L. AGUIAR, A. BARDELAY et al.*	Resonance Parameters and Covariance Evaluations for the Gadolinium Isotopes *L. LEAL, N. LECLAIRE, F. FERNEX et al.*	Detection of a Slow Kinetic Criticality Accident by the Radiation Protection Monitoring System *O. RAVAT*	Options for Demonstrating Criticality Safety for Geological Disposal of UK Spent Fuel *L. PAYNE, R.WINSLEY, T. BALDWIN et al.*
Development of an ISO Standard Related to Geometrical Dimensions for Subcriticality Control *G. NÉRON DE SURGY, A. BARDELAY, Y. BLIN et al.*	Neutron Nuclear Data Measurements at GELINA *S. KOPECKY, J. HEYSE, C. PARADELA-DOBARRO et al.*	Criticality Accident Alarm System Analysis using MCNP6.2 Constructive Solid Geometry/ Unstructured Mesh Hybrid *J. ALWIN, J. SPENCER, G. FAILLA*	Derivation of Waste Package Criticality Controls that Ensure the Long-Term Criticality Safety of a UK Geological Disposal Facility *T.W. HICKS, E.K. PHIPPS, S. DOUDOU et al.*
Reprocessing Facility Periodic Safety Review: how Impact of Aging Effects on Geometrically Safe Equipments is Reviewed *Y. BLIN, G. NÉRON DE SURGY, B. CHÉCIAK et al.*	Improving Nuclear Data Library Predictability by Accounting for Temperature Effects Using Resonance Parameters *I. MEYER, V. SOBES, B. FORGET*	MAVRIC-Scale Sequence for Criticality Alarm System Applications *C. PAREDES-HAYA, E. ESCANDÓN-ORTÍZ, J. LÓPEZ-MÁRQUEZ et al.*	A Generic Criticality Safety Assessment for the Geological Disposal of Wastes Packaged in Shielded Containers *R.A. HOUGHTON, E.K. PHIPPS, T.W. HICKS et al.*
Claims Arguments Evidence *S. GAN, J.A. RYAN*	Development of a Generalized Lattice Symmetry Formulation for Thermal Scattering Law Analysis *N. SORRELL, A.I. HAWARI*	The CAAS-3S Next-Generation Criticality Accident Alarm System *S. PHILIPS, A. GALLOZZI ULMANN, N. HOUFFLAIN et al.*	ANDRA's Post Closure Nuclear Criticality Safety Assessment towards the Licensing Application for CIGEO *C. LOPEZ, M. RALLIER DU BATY, S. SOULET*
Use of Barrier Assessment in Criticality Fault Analysis *L. WHITELEY*	Progress of ^{140,142}Ce Neutron Cross Section Resolved Resonance Region Evaluations *C.W. CHAPMAN, M.T. PIGNI, K. GUBER*	Presentation on the Future Criticality Incident Detection System at AWE *S. GARBETT*	The Credibility of Post-Closure Criticality: Considerations for MOX Spent Fuel and Wastes Containing Uranium-233 at Disposal or from Ingrowth *R. MASON, T. HICKS, L. PAYNE et al.*

16h10 / 17h50

Poster Session

(with Coffee Break)

-1 **POSTERS** *Area*

End of Sessions

WEDNESDAY, SEPTEMBER 18

9h00/10h40

Session 1	Session 2	Session 3	Session 4
-2 Room A - B	-2 Room C - D	-3 Room 1	-3 Conference Room LOUIS ARMAND
Track 5 STANDARDS, ASSESSMENT METHODOLOGY, REGULATIONS	**Track 3** UNCERTAINTY AND SENSITIVITY ANALYSIS	**Track 1** CODES AND OTHER CALCULATION METHODS	**Track 4** MEASUREMENTS, EXPERIMENTS AND BENCHMARKS
Regulating Criticality Safety: The Effect of Temperature on Reactivity *A.J. NICHOLS*	Impact of Covariances between Criticality Benchmarks Experiments on Licensing *A. HOEFER, O. BUSS*	Development of Supercritical Transient Mik Code and its Application to Godiva Core *T. OBARA, D. TUYA*	The Sandia Critical Experiments Program - What Are We Doing for You Now? *G.A. HARMS, D.E. AMES, J.T. FORD et al.*
Regulating Criticality Safety: Use of Burn-Up Credit in the Assessment of Criticality Risk *E. FLANNERY, W. DARBY*	Correlation of HST-001 due to uncertain technical parameters - Comparison of results from SUnCISTT, SAMPLER and DICE *W.J. MARSHALL, F. SOMMER, M. STUKE*	Employment of the Single Eigenvalue Monte Carlo Technique to some Criticality Safety Problems; Comparison with a Standard, Mixed Deterministic - Monte Carlo Approach *K. W. BURN, P. C. CAMPRINI, M. DULUC*	Neutronic Design of Basic Cores of the New STACY *K. IZAWA, J. ISHII, T. OKUBO et al.*
Implementation of Fission Products Credit for PWR MOX *A. COULAUD, Y. BLIN, G. GRASSI*	The Influence of Changes in Nuclear Covariance Data on the Calculation of Ck hor Highly Enriched Uranium Solution Systems *J. CLARITY, W.J. MARSHALL*	The High-Speed Statistical Criticality Evaluation Method Based on the Multidimensional Interpolation for On-Demand Criticality Risk Evaluation *R. KIMURA, Y. HAYASHI*	Improvements in Void Reactivity Worth Measurements Using a Pressure Sensor *J. GODA, T. GROVE, G. MCKENZIE*
The OXNIT Density Law in CRISTAL Package: an Easy Way to Predict the Composition of Dissolved Oxide in Nitrate Solutions *N. LECLAIRE, F. FERNEX, A. BARDELAY et al.*	UACSA Benchmark Phase IV: Role of Integral Experiment Covariance Data for Criticality Safety Validation, Summary of Results *M. STUKE, A. HOEFER, O. BUSS et al.*		Thermal Epithermal eXperiments (TEX): Test Bed Assemblies for Efficient Generation of Integral Benchmarks *C.M. PERCHER, A.J. NELSON, W.J. ZYWIEC, et al.*

Coffee Break

11h10/12h50

Track 5 STANDARDS, ASSESSMENT METHODOLOGY, REGULATIONS	**Track 3** UNCERTAINTY AND SENSITIVITY ANALYSIS	**Track 9** CRITICALITY ACCIDENTS AND INCIDENTS	**Track 4** MEASUREMENTS, EXPERIMENTS AND BENCHMARKS
Overview and Status of Domestic and International Standards for Nuclear Criticality Safety *D.G. BOWEN*	Assessment of Normality for Criticality Safety Bias and Bias Uncertainty Calculation *J. CLARITY, W.J. MARSHALL*	Lessons Learned from the Accumulation of Uranium in a Gas Purification System *L. WETZEL, B. O'DONNELL, T. LOTZ et al.*	Titanium and Aluminum Sleeve Experiments in Water Moderated 4.31% Enriched UO2 Fuel Element Lattices *D. AMES, G.A. HARMS, J.T FORD et al.*
GRS Handbook on Criticality - New Publication in 2019 *F. SOMMER*	Comparing the Whisper Validation Methodology with Machine Learning Methods *P.A. GRECHANUK, M. RISING, T.S. PALMER*	Criticality Safety Aspects of the «Bump Latch» Event At Dungeness B *J.S. MARTIN, D. PUTLEY, M. HENDERSON*	Design Methodology for Fuel Debris Experiment in the New STACY Facility *S. GUNJI, J.B. CLAVEL, K. TONOIKE et al.*
Current Status of Nuclear Regulation in Japan - Focusing on Nuclear Criticality Safety *K. NAKAJIMA*	A Proportionate Approach to EPD *B. PHILPOTTS*	Criticality Incident Detection Decision Making: the Evaluation of Unforeseen Risk *N. HARRIS*	Solution Critical Experiments Partially Reflected by Lucite *M.L. ZERKLE, S.N. BAUER*
Feedback from IAEA TRANSSC Working Group and Technical Expert Group on Criticality *M. MILIN, D. MENNERDAHL, B. DESNOYERS et al.*	Monte Carlo Uncertainty Analysis Method in «Gadolinium Credit» Applications to BWR Cask Configurations *M.CHERNYKH, S. TITTELBACH, J.C. NEUBER et al.*	Criticality Accidents Detection and Minimum Accident of Concern: Review and Discussions *M. DULUC*	Warm Critical Runs in Support of the Kilopower Reactor Using Stirling TechnologY (KRUSTY) Experiment *R. KIMPLAND, R. SANCHEZ*

Lunch

WEDNESDAY, SEPTEMBER 18

14h00 / 15h40

Session 1 (-2 Room A - B)	Session 2 (-2 Room C - D)	Session 3 (-3 Room 1)	Session 4 (-3 Conference Room LOUIS ARMAND)
Track 5 STANDARDS, ASSESSMENT METHODOLOGY, REGULATIONS	**Track 10** PROFESSIONAL DEVELOPMENT ISSUES AND TRAINING	**Track 9** CRITICALITY ACCIDENTS AND INCIDENTS	**Track 4** MEASUREMENTS, EXPERIMENTS AND BENCHMARKS
IRSN Approach for Criticality Accident Assessment *A. BARDELAY, M. DULUC, J. RANNOU*	Renewal of IRSN Training in Nuclear Criticality Safety *C. LENEPVEU, M. DULUC, M.P. VERAN VIGUIE et al.*	Assessment of Re-Criticality in Severe Accident Configurations Using MCNP and MELCOR *M.P. FONTAINE, T. HELMAN, I. MALKINE*	History and Future of Temperature Reactivity Experiments at VR-1 Reactor *T. BILY, L. SKLENKA, F. FEJT et al.*
The New Version of the Criticality Safety Guide Sheets Collection *A. DORVAL, M. PRIGNIAU, P. CASOLI et al.*	Maintaining NCS Capability, Capacity and Competence after Enormous Attrition *N. GLAZENER, J. KUROPATWINSKI, W. CROOKS et al.*	Experience in Evaluations of Criticality Immediately after Accidents with the Destruction and Melting of Nuclear Fuel at NPP *V.V. TEBIN, A.N. BEZBORODOV, A.E. BORISENKOV et al.*	The Effect of Temperature on the Neutron Multiplication Factor for PWR Fuel Assemblies *S. GAN, A.R. WILSON*
Use of ANSI/ANS 8.6 Standard for Criticality Safety Applications in the Modern World of Advanced Simulation Capabilities *W. MYERS, J. ALWIN, N. CHISLER et al.*	Current Status of the DOE/ NNSA Nuclear Criticality Safety Program Hands-On Criticality Safety Training Courses *D.G. BOWEN*	Numerical Analysis of Criticality of Fuel Debris Falling in Water by Combining Computational Fluid Dynamics and the Continuous Energy Monte Carlo Code *M. TAKESHI, J. NISHIYAMA, T. OBARA*	Use of BWR Cold Critical Benchmarks for Code Validation *A. RANTA-AHO*
Extensive Study of the Heterogenous Repartition of the Moderation when both the Fissile Mass and the Moderation are Controlled *M. DULUC, J. HERTH, F.X. LE DAUPHIN et al.*	University Pipeline Program for the Education of Future Nuclear Criticality Safety Professionals *J. MCCALLUM, A. MEREDITH, J. BUNSEN*	Exploratory Investigation for Estimation of Fuel Debris Criticality Risk *Y. YAMANE, Y. NUMATA, K. TONOIKE*	Steady-State Benchmark Evaluation of the TREAT M2 and M3 Calibration Experiments *N.C. SORRELL, A.I. HAWARI*

Coffee Break

16h10 / 17h50

Track 11 FUTURE CHALLENGES	**Track 10** PROFESSIONAL DEVELOPMENT ISSUES AND TRAINING	**Track 9** CRITICALITY ACCIDENTS AND INCIDENTS	**Track 4** MEASUREMENTS, EXPERIMENTS AND BENCHMARKS
Status of the NEA International Activities on Nuclear Criticality Safety *S. TSUDA, F. MICHEL-SENDIS, T. IVANOVA et al.*	Criticality Safety Training at CEA *M. PRIGNIAU, E. FILLASTRE, F. LESPINASSE et al.*	Supercritical Kinetic Analysis in a Simple Fuel Debris System by MIK code *K. FUKUDA, D. TUYA, J. NISHIYAMA et al.*	Criticality Testing of Recent Measurements at the National Criticality Experiments Research Center *J. HUTCHINSON, J. ALWIN, R. BAHRAN et al.*
An Overview of the United States Department of Energy's Nuclear Criticality Safety Program and Future Challenges *D.G. BOWEN, A.S. CHAMBERS*	«Criticality Safety Analysis» Training Course for Engineers to Be Qualified in Criticality Safety *A. DORVAL, D. NOYELLES, M. PRIGNIAU et al.*	Multiphysics Coupling Analysis for Spent Fuel Pool Loss of Coolant Accident *J.A. BLANCO, P. RUBIOLO, E. DUMONTEIL*	Validation of New Silicon Evaluation in Special Core of LR-0 Reactor *T. CZAKOJ, M. KOŠŤÁL, E. LOSA et al.*
Future Challenges in Re-Establishing a Solution Critical Capability in the United States *C. PERCHER, D. HEINRICHS, S. BATES et al.*	Criticality Training for the Active Handling Facility *J. RENDELL*	Multiphysics Simulation of Two Criticality Accident Excursions in Lady Godiva Using MCATK *T.J. TRAHAN, S. DOSSA, R.H. KIMPLAND et al.*	Benchmark Evaluation of Saxton Plutonium Program UO2-Fueled Critical Lattices *B. SAENZ, M.A. MARSHALL, J.D. BESS*
Nuclear Criticality Safety Beyond 2019 *D.K. HAYES*	Training for Fissile Material Handlers, Supervisors, and General Personnel *Q. BEAULIEU, J. BUNSEN*	Criticality Accident Safety Analysis: Questions and Partial Answers Provided by Dedicated Experiments Conducted on CRAC and SILENE *F. BARBRY, M. LAGET, M. PRIGNIAU*	Investigation of the Impact of the Prediction Error of the Burn-Up Code System SWAT4.0 on Neutronics Calculation *K. TADA, T. SAKINO*

End of Sessions

THURSDAY, SEPTEMBER 19

9h00/10h40

Session 1	Session 2	Session 3	Session 4
-2 Room A - B	-2 Room C - D	-3 Room 1	-3 Conference Room LOUIS ARMAND
Track 11 FUTURE CHALLENGES	**Track 10** PROFESSIONAL DEVELOPMENT ISSUES AND TRAINING	**Track 9** CRITICALITY ACCIDENTS AND INCIDENTS	**Track 4** MEASUREMENTS, EXPERIMENTS AND BENCHMARKS
Progress of Criticality Control Study on Fuel Debris by Japan Atomic Energy Agency to Support Secretariat of Nuclear Regulation Authority *K. TONOIKE, T. WATANABE, S. GUNJI et al.*	Nuclear Criticality Safety Training at the National Criticality Experiments Research Center *D.K. HAYES*	Criticality Accident Phenomenology: Numerical Experiments as a Learning Tool *M. LAGET*	Analysis of the Criticality Benchmark Experiments Utilizing UO2F2 Aqueous Solution in Spherical Geometry *T. GORICANEC, B. KOS, G. ŽEROVNIK et al.*
Nuclear Criticality Safety Impacts of Additive Manufacturing *K. WESSELS, M. KNOWLES*	Education and Training at VR-1 Reactor Facility. Can Be Benefiting for Criticality Safety Engineers? *T. BILY*	Review of IRSN Work Regarding Nuclear Criticality Accident *M. DULUC, J. RANNOU, F. TROMPIER et al.*	Criticality Analysis of NCA Critical Experiments Simulating SFP of Low Moderator Density Conditions *S. SHIBA, D. IWAHASHI*
Criticality Characteristics of Fuel Debris Mixed by Fuels with Different Burnups Based on the Fuel Loading Pattern *T. WATANABE, K. OHKUBO, S. ARAKI et al.*	Criticality Augmented Reality Training Aid *S. HAY, T. PAGE, C. HOLLAND et al.*	Impact of Criticality Accident Characteristics on Sellafield Criticality Emergency Arrangements *D. KIRKWOOD, A. WILSON, C. CUMMING*	Detailed Design of Epithermal/ Intermediate Spectrum Critical Experiment Using the Sandia National Laboratories Critical Facility *J. CLARITY, T. MILLER, W.J. MARSHALL et al.*
Application of the Neutronic Part of the Nuclear Simulation Chain of GRS to Accident Tolerant Fuel systems - First Results *R. KILGER, R. HENRY*	Safety Analysis Report for Packaging (SARP) Shielding & Nuclear Criticality Safety Generalist and Analyst Courses Developed and Conducted by Oak Ridge National Laboratory *D.G. BOWEN, J. RISNER, G. RADULESCU et al.*	Needs and State of the Art in Criticality Dosimetry and Dose Reconstruction Techniques for Medical Management of Criticality Accident's Casualties *F. TROMPIER, M.A. CHEVALIER*	Results of Newly Expanded COG Criticality Validation Suite *D.P. HEINRICHS, S. KIM*

Coffee Break

11h10/12h40

Closing Ceremony

-3 *Conference Room* ***LOUIS ARMAND***

12h40/13h00

Departure for the Technical Tours

-1 ***RECEPTION DESK***

Workshops

14h00/15h20

-3 Room 3	-3 Room 4
Workshop 1 Second Level Criticality Modelling with CRISTAL Package: Enhancing Criticality Safety Assessments for Industrial Applications *Y. RICHET et al.*	**Workshop 2** Enhancing Validation of Nuclear Criticality safety Calculations with ICSBEP Handbook an NEA Tools *J. BESS, I. HILL, S. TSUDA*

Coffee Break

15h40/17h00

Workshop 1 Second Level Criticality Modelling with CRISTAL Package: Enhancing Criticality Safety Assessments for Industrial Applications *Y. RICHET et al.*	**Workshop 2** Enhancing Validation of Nuclear Criticality safety Calculations with ICSBEP Handbook an NEA Tools *J. BESS, I. HILL, S. TSUDA*

End of Sessions

Abstracts

MONDAY, SEPTEMBER 16

Session 1 > Room A-B

12h00 - 12h50 > Track 6

THE DISPOSAL OF THE FINAL CONCENTRATE MOTHER LIQUOR FROM THORP: HOW CRITICALITY SAFETY ANALYSIS CAN IMPACT OVERALL RISK

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On the Sellafield site, the Thorp (THermal Oxide Reprocessing Plant) reprocesses oxide spent fuel from British Advanced Gas Cooled Reactors and Light Water Reactor fuel from foreign sources. It achieves this by dissolving sheared fuel and separating the fission products, uranium and plutonium via solvent extraction. Purification of the product liquor and conversion into powder for long term storage is then undertaken. A by-product of the conversion process is the generation of plutonium-bearing Concentrate Mother Liquor (CML), which is reworked to the start of the solvent extraction process. Thorp has recently completed its commercial reprocessing operations and now the entire plant is undergoing a period of rundown and washout to enable Post Operational Clean Out (POCO). Due to this rundown and washout phase, the final CML from the last plutonium liquor cannot be reworked back into the solvent extraction process as it would be under commercial operations. As one of the prime directives of POCO is to prevent any orphan wastes remaining in the facility an alternative disposal route for this final batch of CML is required.

An in-depth optioneering process has been undertaken and such routes have been identified. However, all of these routes carry a non-negligible criticality risk. For example, one of the routes involves sending the CML to a downstream effluent waste plant which primarily receives highly active raffinate feeds that have a very low fissile content. These feeds that are evaporated and the resultant liquor vitrified into a glass medium that is suitable for long term storage. This effluent plant can receive non-negligible amounts of solvent. Solvent can extract fissile material such that unsafe fissile concentrations are possible. Limiting the volume of solvent received into the effluent plant from other sources would help to reduce the amount of criticality risk that this disposal route would carry. In order to facilitate such a removal the other feeds into this downstream plant would have to be embargoed. The outcome of this would be a potentially intolerable drop-off in the quality of the finished waste product (vitrified glass) such that the ability of the glass to meet the stringent requirements for geological disposal would be questioned.

This paper will examine how decisions to lower the criticality risk can have a potentially adverse impact on plant operations when assessed in the context of overall risk reduction and what measures were put in place to ensure that the CML disposal met the UK legal requirement of demonstrating the risk to be As Low As Reasonably Practicable (ALARP).

RECENT IMPROVEMENTS IN THE K-AREA CRITICALITY SAFETY PROGRAM

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The K-Area facility at the Savannah River Site provides for the handling and interim storage of the United States' excess plutonium. Operations consist of plutonium storage in large arrays of shipping packages and surveillance capabilties that include various non-destructive analysis instruments and one glovebox. Through a philsophical shift and many technical improvements in the criticality safety documentation, the number of required controls has been significantly decreased.

Up through 2016, a suite of criticality safety evaluations was in place for all of K-Area's fissile material operations. These nine evaluations established requirements for 315 credited criticality safety controls. Among these evaluations, there was a high

reliance on quantitative frequency analysis to demonstrate that the frequency of a criticality accident was less than 1E-6/year. These frequency analyses were used as justification that a criticality accident alarm system was not needed. The large number of controls was a result of using quantitative frequency analysis as well as performing very detailed computational analyses that implied many details were important to criticality safety and must be controlled.

Starting in 2014, a philosophical shift began to enable a change in approach for criticality safety in K-Area. The old approach determined how many controls were needed to quantitatively prove that the frequency of a certain upset event sequence would remain below 1E-6/year. The new approach determines which controls are needed to ensure that credible upset conditions will remain subcritical. During the same time, a new methodology was developed for assessing the need (or lack thereof) for a criticality accident alarm system [1]. Instead of relying on quantitative frequency arguments, the new methodology asseses the aggregate risk of a criticality accident in the facility based on the nature of the fissile material processes and the complexity of operations, among other factors. This change in methodology enabled the abandonment of quantitative frequency analyses.

In the past five years, the nine criticality safety evaluations have been revised, consolidated, or replaced altogether with new evaluations. These new evaluations employ more reliance on American Nuclear Society standards, handbook values, and hand calculation methods. These new evaluations rely less on Monte Carlo calculations and do not rely at all on quantitative frequency calculations. The result is five evaluations that contain a total of 100 controls, which is a reduction of 68%. At the same time, operational capability, throughput, and efficiency have been increased.

In this paper, the philosophical change and technical improvements will be presented, and the resulting elimination of controls will be discussed.

[1] J.S. Baker, et al, "Assessment of the Need for a Criticality Accident Alarm System," Transactions of the American Nuclear Society, Volume 111, pp.850-853 (2014).

14h00 - 15h40 > Track 6

THE BENEFIT OF DIFFERENT APPROACHES TO BOUNDING POISON QUANTIFICATION

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A licensee has several tanks containing uranyl nitrate liquors from past processing operations. The liquors have been poisoned with gadolinium nitrate, intended to ensure the stored quantity of material is sub-critical even in spherical geometry in anticipation of future processing, storage or disposal requirements. The tanks contain liquor of different enrichments, different uranium (U) concentrations and different gadolinium (Gd) concentrations. Prior to further processing the liquors need to be blended to produce a reasonably uniform liquor.

It was decided to produce the safety assessment without taking credit for geometry but taking credit for the presence of the Gd. This will provide the necessary confidence that future operations could be completed safety.

The volume of the liquor is such that only a small proportion of the total would fit in a single tank. The blending operations are therefore a complex series of small transfers to provide reasonably uniform enrichment and concentration. An error in the transfers was determined to be reasonably likely and so the assessment needed to ensure that any intermediate mixing state would remain sub-critical.

To bound any intermediate mixing state it was decided to consider a combined material consisting of the highest ^{235}U concentration present in any tank, the highest enrichment present in any tank and the lowest Gd concentration present in any tank. In this way no intermediate mixing state could produce a material more reactive than the combined material. Faults such as evaporative losses and mixing with water from other sources e.g. from firefighting also needed to be considered. The combined material was taken as the starting point, but the U concentration was varied by removing or adding water while conserving the mass of Gd and U. This was modelled as spherical geometry with full water reflection. The results *(Figure 1)* show that at 92% enrichment, 15 kg of ^{235}U remains sub-critical at optimum concentration. However, the above approach gave some insight that greater margins of safety could be demonstrated. Inspection of the tank contents showed that low Gd concentrations generally occurred when the ^{235}U concentration was also low. The combined material described above is, therefore, highly pessimistic. It was determined that the lowest Gd:^{235}U atomic ratio within any tank was in excess of 0.04. Hence, no intermediate mixing state could reduce the Gd:^{235}U atomic ratio further. This was modelled as above but at 100% enrichment and varying the ^{235}U concentration but maintaining the Gd:^{235}U atomic ratio at 0.039. The results from this model *(Figure 2)* show that up to 24.5 kg of ^{235}U remains sub-critical under optimum conditions. This approach, therefore, enabled a relatively simple safety assessment to be completed while determining larger safety margins.

Figure 1. k-eff v ^{235}U Concentration for 15 kg of ^{235}U with fixed Gd and U Starting Conditions, 230g (^{235}U)/l and 3.1 g(Gd)/l.

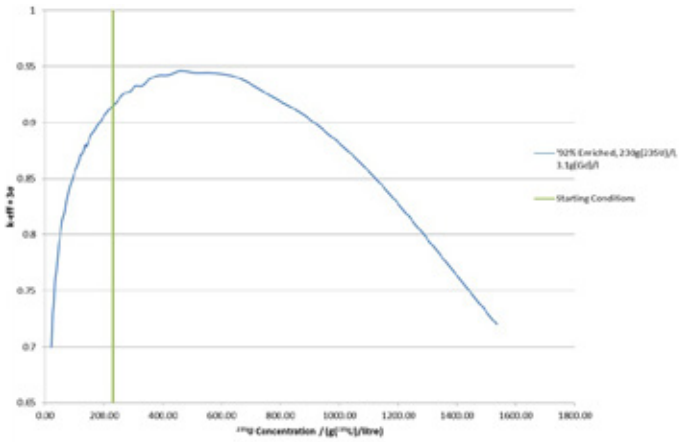
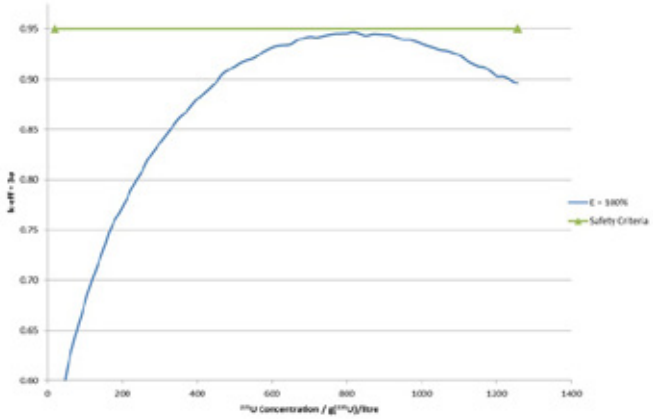


Figure 2. k-eff v ^{235}U Concentration for 24.5 kg ^{235}U and a fixed Gd: ^{235}U ratio of 0.039.



CLEANING AND DISMANTLING OF HOT CELLS DEDICATED TO MECANICHAL TREATMENT AND SHEARING OF SPENT FUEL – CRITICALITY SAFETY ISSUES

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The Marcoule Pilot Plant ("Atelier Pilote de Marcoule" - APM) was a semi-industrial facility dedicated to the reprocessing of spent fuel. It was used to validate the reprocessing flowsheets for natural uranium fuels, then fast neutron reactor fuels and light water reactor fuels. Since 1997, operation has been stopped and the plant is being cleaned up and dismantled. APM includes three nuclear buildings (211, 213, and 214). Building 214, operated from 1988 to 1997, was used for the reception, storage, mechanical treatment and dissolution of spent fuel, as well as clarification of dissolution solutions. This building includes hot cells 418 and 421, which are dry hot cells, where fuel cases were opened and fuel pins were sheared. The remaining fissile materials in these cells are mostly powders accumulated during operation, mainly in Cell 418.

The preparatory operations for dismantling can be divided into two main steps:

- First, equipment cleaning and disassembly were carried out, with no significant modification of the criticality safety case,
- Then, dismantling operations, with the cutting of process equipment and wet cleaning were initiated. These operations required the development of a new criticality safety case.

The aim of this paper is to present the issues associated with the new criticality safety case with regard to this second step, including:

- The evaluation of the masses of fissile material initially remaining in the cells, and removed in the waste produced during dismantling,
- The choice of criticality control modes, taking into account the progress of the operations: limitation of mass and moderation, then limitation of the mass only (no moderation limit),
- The choice of the fissile reference medium, which required a specific justification given the various types of fuels that have successively been treated in the cells,
- The criticality calculations carried out and the criticality safety case, which includes a follow-up of the recovered masses of fissile material, with associated limits and hold points.

In addition concerning criticality safety, feedback from the dismantling operations carried out to date will be presented. In particular, the evaluation of the masses of fissile material actually recovered will be compared with the limits associated with hold points, and with the maximum permissible limits for subcriticality.

HOMOGENISATION TECHNIQUES FOR BOUNDING CRITICALITY SAFETY ANALYSES FOR FUEL FABRICATION AND REPAIR

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We present an investigation of homogenisation techniques in criticality safety analyses of partly assembled fuel rod lattices as they appear in connection with fuel fabrication and repair. These techniques allow us to continuously and uniformly vary the moderator-to-fuel ratio over defined zones of the fuel assembly in order to tune the moderation state to optimum moderation leading to maximum reactivity. Two different homogenisation methods are compared. The first method involves computation of cell-weighted cross sections of the empty lattice cells followed by homogenising the material of the occupied and unoccupied lattice cells in the considered lattice zone. For the second method, the moderator-to-fuel ratio is tuned to the value defined by the number of occupied and unoccupied lattice positions by adapting the pitch of the fuel lattice accordingly. Comparing the results obtained with the two homogenisation methods to the results obtained for discretely modelled fuel lattice configurations demonstrates that both homogenisation methods are suitable for bounding criticality safety analyses of partly assembled fuel

assemblies. Models with two zones prove to be sufficient in this context, where the outer lattice positions are occupied with fuel rods to minimise neutron leakage, and the interior of the fuel assembly is tuned to a state of optimum moderation. Since the homogenisation methods are easy to use and minimise the analysis effort, they are well-suited for criticality safety analyses of partly assembled fuel rod lattices.

A SIMPLE ALTERNATIVE APPROACH FOR THE MODELLING OF FUEL ASSEMBLIES WITH MISSING FUEL RODS

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This paper presents a discussion about an alternative calculation scheme that can be used to model nuclear fuel assemblies under water.

In criticality Monte-Carlo multigroup calculations performed by French industrials, heterogeneous media (such as fuel assemblies or more generally any type of arrays of fissile material in water) are mostly modelled as a homogeneous medium whose nuclear cross sections are adjusted to match those of the original heterogeneous system. These cross sections are obtained using a preliminary flux calculation based on deterministic methods. For a fuel assembly, the simplest deterministic method is to consider a simple cell (fuel oxide cylinder surrounded by clad and water) with a moderation ratio determined by considering that the water around fuel rods and inside empty slots is homogeneously distributed all over the assembly section. This simplified calculation scheme, very useful for parametric studies and for modelling missing fuel rods which position is not known, allows fast multigroup calculations and only requires knowing the number of fuel rods in the section. However, in some cases, this simplified approach can lead to underestimate the assembly reactivity. Indeed, the moderation ratio is generally heterogeneous within the assembly section, particularly for boiling water reactor types assemblies for which empty slots are unequally dispatched within their section. Moreover, the modelling of the water located outside of the assembly section can lead to mis-estimate the moderation of the external fuel rods.

The purpose of this paper is to present an alternative simple approach to use preliminary deterministic calculations to generate homogeneous cross sections for multigroup Monte Carlo codes, in order to avoid two dimensional deterministic calculations which limit the parametric studies possibilities and are computer-time consuming.

This paper will firstly remind generalities about the problematics regarding missing fuel rods in fuel assembly modelling. Then, results obtained with the alternative calculation scheme will be presented, with an explanation of the discrepancies compared to other calculation schemes. At last, a discussion about the bounding trait of this alternative scheme and the parameters that can have an influence on its behaviour will be presented.

16h10 - 17h50 > Track 6

INTEGRATION OF UNCERTAINTIES INTO THE SAFETY ANALYSIS FOR A LARGE NUMBER OF MOVEMENTS

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For mass-controled workstations, the evaluation of the mass of fissile material in the gloveboxes is carried out by means of a mass balance: mass measurement and balance of inputs and outputs. Mass balance usually does not integrate uncertainties. The establishment of an internal threshold, lower than the mass limit defined by the workstation safety analysis, ensures that the mass limit is not exceeded at any time. In the case of worksation affected by a large number of movements of fissile material, a trivial way to include the uncertainties (of balance type = mass input + Delta_input - mass output + Delta_output) leads to an impossibility to define a threshold of positive value. However, the application of the method of propagation of independent uncertainties makes it possible, while ensuring a pre-determined and constant confidence level regardless of the number of movements, to set a value of the threshold that is compatible with an industrial operation. This method is presented in this paper.

NUCLEAR CRITICALITY SAFETY ASSESMENT SUPPORTING THE INTEGRATED SAFETY ANALYSIS OF THE PELLET FABRICATION PROCESS AT THE JUZBADO PLANT

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The Integrated Safety Analysis (ISA) carried out at the Juzbado Fuel Fabrication Plant aims to identify all the potential accident sequences that might occur during the plant's lifetime. During the manufacturing process of green pellets, and particularly on the blending and homogenization stages, where hydrogenated additives are involved, nuclear criticality safety focuses on neutron moderation, which is controlled by limiting the Hydrogen – Uranium atomic ratio or H/U. The ISA has identified potential accident sequences that challenge the H/U safe-limit value, both for uniform and non-uniform over-moderation scenarios.

Three accident sequences have been addressed, the so called Less Uranium Powder, Process Improperly Done and More Hydrogenated Additive scenarios. These three sequences lead to exceeding the H/U limit value and so, we analyzed in detail how reactivity behaves during them in order to achieve better understanding of the fabrication process from a criticality safety point of view.

In this way, the 'Less Uranium Powder' sequence considers a fixed amount of hydrogenated additive, homogeneously mixed with a decreasing mass of uranium. Although the H/U safe-limit value is exceeded, we see that reactivity slumps as uranium mass decreases.

Regarding the 'Process Improperly Done' sequence, it is assumed that hydrogenated additive is not homogeneously mixed with uranium, giving place to an over-moderated region inside the equipment. This region is modelled as a hemi-spherical shaped region where uranium is increasingly added, reaching a homogeneous mixture at the end of the sequence. Under these conditions, we see that during the process the system remains subcritical, but the reactivity increases from a certain starting value and so, the safety margin is decreased.

On the contrary, the sequence 'More Hydrogenated Additive' considers an increasing mass of hydrogenated additive, homogeneously mixed with the uranium. This sequence not only exceeds the H/U limit value, but also makes the system go critical, which allows us to obtain the maximum additive mass ensuring Subcriticality.

Finally, according to the ISA methodology, we were compelled to implement IROFS to ensure that the process complies with the performance requirements established by the regulations. One of those IROFS was developed from the conclusions of the aforementioned safety assessment, and consists of a device that acts as a barrier to prevent the occurrence of the 'More Hydrogenated Additive' and 'Process Improperly Done' sequences. This system firstly limits the mass of additive that can be poured on the equipment by means of a passive-engineered control (pre-set volume container) and secondly by preventing the pouring of a second additive-container into the blend through an active-engineered control. Besides, the device implements an alveolar valve, ensuring that the pouring of additive is fractioned over time through a dosage/time preset program.

URANIUM ACCUMULATIONS IN CASTING OPERATIONS

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In December 2017, unexpected accumulations of uranium were discovered in several locations throughout the casting process area at the Y-12 National Security Complex that subsequently placed casting operations on hold. The Y-12 uranium casting process takes place in enclosures and hoods, many of which are open to a large floor surface area that has limited visibility and access. At the time of discovery, there were no Nuclear Criticality Safety (NCS) analyses or limits for material accumulations in the casting processing areas. Due to changes in the operation of the casting process and invalid assumptions in the criticality safety evaluation, significant material accumulations were occurring inside the enclosures. The material accumulations were identified as a result of an extent of condition review from another process, and an occurrence report was filed. The NCS organization worked with Production and Operations personnel to identify additional areas of uranium accumulation in the casting area, which resulted in several iterations of cleaning and material collection. Following material recovery, a causal analysis was performed that identified several contributing factors that allowed the accumulations to go unnoticed. In order to resume operations, the criticality safety evaluation was revised to analyze and control uranium accumulations in the casting area. Additionally, improvements in nuclear material control and accountability (NMC&A) controls were required. Casting operations have since resumed with frequently required material collections, which have limited the amount of uranium material accumulation.

MANAGING THE RISK FROM FLOODING FOR A FACILITY AT AWE

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Withstand to natural external hazards is an area of interest for any building containing significant quantities of fissile material. One such hazard is extreme rainfall resulting in topographic flooding that may present a criticality safety hazard.

The subject of this paper is a facility on the UK's Atomic Weapons Establishment (AWE) site. The first Periodic Review of Safety (PRS) for the facility utilised evidence from historic flooding in combination with rainfall data from the UK's National Meteorological Service to determine the design basis challenge for the facility in accordance with guidance at the time. Relatively minor modifications to protect fissile material ensued and the risk from criticality was deemed negligible. More recently, a flooding event affecting another facility and the surrounding area prompted new site wide flood modelling. This new modelling resulted in a revision to the design basis flooding challenge for the facility. This modelling also took account of the release of a new national reference source for rainfall data in the UK. The bounding topographic flooding challenge was identified to arise from a short duration summer storm and resulted in a significant increase in the predicted flood depth, over and above that considered in the first PRS.

This paper presents an overview of the response to the change in predicted topographic flood depth for the facility on the AWE site, and how the risk from criticality due to flooding is being managed. The paper describes the considerations made when presented with the results of the revised data and the subsequent sequence of events, including: the derivation of the estimated flood depths, including the application of uncertainties and margins to this data; identification of vulnerable locations within the facility; optioneering for identification of immediate and long term fixes to mitigate/eliminate the flooding risk in these areas (taking account of environmental and radiological

consequences in addition to criticality); construction of enduring civil works to protect against flooding for the areas of greatest concern; on-going assessment and the implementation of local improvements to support future operations.

Session 2 > Room C-D

12h00 - 12h50 > Track 2

DEVELOPMENT AND IMPLEMENTATION OF AN IMPROVED LIQUID TSL TREATMENT IN THE FLASSH CODE

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Thermal scattering law (TSL) analysis of liquid materials emerges in various applications of nuclear science and engineering. These include both traditional light water reactors and advanced thermal reactors, where moderators such as liquid FLiBe may be a major constituent of the core. To describe the phenomenon of neutron thermalization in liquid neutron moderators, typical TSL analyses (e.g., using the NJOY code system) assume that the TSL is comprised of a primary, solid-like component and a secondary component accounting for the diffusional behavior prevalent in liquids. In this work, a module that treats liquid physics, specifically diffusional phenomena, has been implemented in the Full Law Analysis Scattering System Hub (FLASSH) nuclear data code (developed by LEIP Labs at North Carolina State University). The new module contains all the previous capabilities that utilize free gas and Schofield diffusion models. However, it also introduces alternative diffusion models (e.g., the Langevin model for high viscosity fluids), physics-based gridding schemes for capturing the complete liquid TSL with appropriate resolution, a more general convolution algorithm for combining the solid and liquid components in a more straightforward manner, and a flexible, modular framework for adding more advanced features in the future (e.g., automatic alpha-beta grid generation for liquid systems). The operation of this module was tested to reproduce the results of past TSL evaluations (i.e., light water and heavy paraffinic oil) and to investigate the diffusional component of previously unevaluated materials.

MEASUREMENT OF THE DOUBLE-DIFFERENTIAL NEUTRON CROSS SECTION OF UO_2 FROM ROOM TEMPERATURE TO HOTT FULL POWER CONDITIONS

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Phonon densities of states (PDOS) of U in UO2 and O in UO2 have been determined from double-differential neutron cross sections of UO_2 measured at the Institute Laue Langevin (Grenoble) at 300 K, 600 K and 900 K. The obtained PDOS were used to generate temperature-dependent thermal scattering laws (TSL) with the LEAPR module of the processing code NJOY. The impact of the TSL on UOX fuel calculations was quantified with the Monte-Carlo code TRIPOLI4â. At Room temperature, the use of TSL of UO_2 in neutronic calculations implies a low decrease of the calculated reactivity ranging from -50 pcm to -100 pcm.

14h00 - 15h40 > Track 2

INTERNATIONAL BENCHMARKS INTERCOMPARISON STUDY FOR CODES AND NUCLEAR DATA VALIDATION

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In collaboration with the Department of Energy (DOE) Nuclear Criticality Safety Program (NCSP), the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) is leading a new benchmark intercomparison based on the MORET Monte Carlo code using various nuclear data libraries (JEFF-3.3, ENDF/B-VII.1 and ENDF/B-VIII.0) and a large selection of benchmarks. Their results are collated with those from Lawrence Livermore National Laboratory (LLNL), Los Alamos National Laboratory (LANL) and Oak Ridge National Laboratory (ORNL) using respectively the COG, MCNP and KENO (SCALE package) Monte Carlo codes

associated with ENDF/B-VII.1 and ENDF/B-VIII.0 libraries. Due to the large number of benchmarks involved (about 2000), this effort is envisioned to take three years and is currently focused on High Enriched Uranium systems (HEU) and Plutonium systems (PU). About 450 HEU and 500 PU benchmarks taken from the ICSBEP handbook were considered covering a large energy spectra range (from thermal to fast) and a wide range of isotopes. Rigorous cross-checking of results using same nuclear data evaluations has already revealed subtle modeling and interpretation user errors, as well as some inconsistencies in the DICE database, that will be gathered and reported to the ICSBEP working group. After resolving and correcting errors in input decks, some small remaining discrepancies indicate possible issues with the processing of nuclear data and thermal scattering law. Once confident in the benchmark modeling, the comparison of the results obtained with various libraries allows validating nuclear data of various isotopes of interest for criticality safety and highlighting the improvement made these last years in the state-of-the art nuclear data libraries.

EVALUATION UPDATES FOR MINOR AND MAJOR ACTINIDES

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Evaluations of most physical observables are based on a statistical analysis of a compilation of experimental data, model calculations and their uncertainties, heavily biased toward the former for observables with sufficient experimental data. Hence, as new experimental data become available, they are incorporated into the evaluation files. The latest LANL measurements at LANSCE of the neutron capture on ^{234,236,238}U, as well as measurements of the fission cross section ratio to ^{235}U for both ^{234}U and ^{236}U have prompted the re-evaluation of all the reactions channels for these reactions. In the case of ^{239}Pu, the evaluation of the prompt fission neutron spectrum (PFNS) is updated based on LANSCE measurements by the Chi-Nu collaboration. In addition to discussing the evaluation of the minor U isotopes, we give an update of the PFNS evaluation, as well as latest cross section evaluations for neutron induced reactions on ^{239}Pu.

TESTING NEW THERMAL SCATTERING LAW FOR LIGHT WATER AT 600 K USING VESTA 2.2 DEPLETION CALCULATIONS

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Recent releases of the US and European evaluated nuclear data libraries, namely ENDF/B-VIII.0 and JEFF-3.3, included H_2O reviewed versions of thermal scattering law (TSL) data. The ENDF/B-VIII.0 adopted evaluation is based on molecular dynamics (MD) simulations whereas JEFF data are based on experimental measurements. While the ENDF and JEFF evaluations perform decently well for Room temperature, performance for temperatures other than Room temperature is needed. At IRSN, an effort has been devoted to the investigation and evaluation of H(H_2O) TSL for temperatures higher than Room temperature for reactor safety and criticality safety applications. For so, thermal neutron scattering cross section for hydrogen bound in light water was re-evaluated at reactor operating temperature, i.e., at 600 K based on MD simulations using the PolarisMD code and TCPE polarizable rigid water model potential. In this work, we present the impact of this new IRSN TSL data for light water for PWR using VESTA 2.2.0 depletion calculations, by comparing the results obtained using both ENDF/B-VIII.0 and the new TSL data. The VESTA 2.2.0 calculations are performed for the ARIANE.GU3 sample, which is a PWR UO_2 fuel rod sample with an estimated burn up of 52.5 MWd.kgHM^{-1}, irradiated in the Gösgen PWR in Switzerland between 1994 and 1997 during three cycles. The results are compared to the ones obtained with previous thermal scattering cross sections for light water from the JEFF evaluations. For validation purpose, they are also compared to experimental data.

PROGRESS ON THE RECONR MODULE FOR NJOY21

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A basic but particularly essential step in the production of nuclear data application libraries is handled by the RECONR module. It is used to reconstruct resonance cross sections from resonance parameters and to reconstruct cross sections from ENDF nonlinear interpolation schemes. Cross sections are then evaluated on a common energy grid and redundant cross sections are determined from their components. The RECONR module is the first module being modernized under the NJOY21 project. In this paper we will provide an overview of the current status of the modernization of this module. In particular, we will present the first results for the new R-Matrix limited (RML) resonance reconstruction capability provided by NJOY21.

16h10 - 17h50 > Track 2

MEASUREMENT OF GAMMA RAYS FROM RADIATIVE CAPTURE OF URANIUM-238 AND DECAY OF SHORT LIVED FISSION PRODUCTS IN SUBCRITICAL SYSTEM

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A subcritical uranium core moderated by polyethylene is mocked up in the Kyoto University Critical Assembly facility. The average ^{235}U enrichments of the core is 5.4 wt%. The core is driven by a ^{252}Cf neutron source and γ ray spectrum is measured with a HP-Ge detector. The radiative capture g rays of 4.060 MeV is observed from ^{238}U in spite that the emission of it is not predicted in a neutron-photon coupled transport simulation with the library based on the nuclear data file JENDL-4.0. Simultaneously, g rays of discrete energy peaks from short lived fission products are measured as well as the fission prompt g rays of the continuum spectrum. The g ray spectrum is considered to include information of the fissile enrichment. Accordingly, identification and quantification of the peaks are examined.

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IMPACT OF EXPERIMENTAL CORRELATION ON TRANSPOSITION METHOD CARRY OUT WITH CRITICAL INTEGRAL EXPERIMENTS

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In order to estimate the bias on the effective multiplication factor (k_{eff}) of a criticality application case, and the associated uncertainty due to Nuclear Data (ND), a method, taking advantage of the integral experiments information and based on ND sensitivity/uncertainty analyses and adjustment, has been implemented in a tool called RIB (Représentativité, Incertitude, Biais). The latter, developed at the CEA, is related to the experimental validation database of the French criticality-safety package, CRISTAL V2.0, containing more than 2000 experiments from the International Criticality Safety Benchmark Evaluation Project handbook (ICSBEP) and French experimental programs. In most cases, even if a correlation is identified between the experiments, the value of this correlation might not be known.

Validation tests of the RIB tool, in particular by applying strong correlations between integral experiments (>0.9), point out some unrealistic results: a strong deviation of the bias with significant reduction of the uncertainty due to ND. To check those results, equations of transposition method has been implemented in Matlab. Another tool developed at CEA and dedicated to ND evaluation, CONRAD (COde for Nuclear Reaction Analysis and Data Assimilation), has been also used to estimate bias and uncertainty due to ND through the transposition method using integral experiments. Results obtain with these three tools from several combinations « Application case/integral experiments », have been compared, confirming the RIB observed tendencies. Focusing on the ND adjustment, it seems that such phenomenon is associated to a strong variation of the cross sections which might be due to Peelle's pertinent puzzle effect. This paper described the different tools and presents the results obtained for the tested combinations « Application case/integral experiments » in function of the experimental correlation factor. Some potential explanations of the observed results, using strong experimental correlation factors, are discussed.

Session 3 > ❸ Room 1

12h00 - 12h50 > Track 1

SOLOMON: A MONTE CARLO SOLVER FOR CRITICALITY SAFETY ANALYSIS

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A new Monte Carlo solver Solomon has been developed for the application to fuel-debris systems. It is designed not only for usual criticality safety analysis but also for criticality calculations of damaged reactor core including fuel debris. This paper describes the current status of Solomon and demonstrates the applications of the randomized Weierstrass function (RWF) model and the RWF model superposed with voxel geometry.

DEVELOPMENT OF TERRENUS, A MULTIPHYSICS CODE FOR SPENT NUCLEAR FUEL CASK CRITICALITY ANALYSIS

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A new multiphysics code, Terrenus, is in development at Oak Ridge National Laboratory (ORNL). The purpose of Terrenus is to simulate the consequences of a spent nuclear fuel (SNF) dual purpose (storage and transportation) canister (DPC) entering a critical configuration while in a geological repository, including calculating heat generation, pressure buildup, and increase in radionuclide inventories.

Subcritical configuration in loaded SNF DPCs is usually maintained by using neutron absorber panels placed between fuel assemblies. These absorbers are typically composed of boron carbide and aluminum. Since repository periods of interest are at least 10,000 years long, it is unlikely that aluminum-based absorbers will maintain criticality control for that amount of time in an aqueous environment. Many currently loaded DPCs may achieve criticality under certain conditions (e.g., loss of absorber panels in a flooded scenario) over a repository time frame. Therefore, criticality consequence analysis is needed to support direct disposal of DPCs. Direct disposal of loaded DPCs has many benefits, including billions of dollars in cost savings.

Two criticality-inducing scenarios are under consideration. The first is a slow approach to critical as subterranean water seeps into an SNF canister with degraded neutron absorbers. The second is a rapid criticality insertion due to some unspecified event such as the sudden collapse of the neutron absorber support structure in a flooded cask. The current development focus is on the first scenario, but ultimately, Terrenus will be used to simulate both scenarios.

The simulation software suite will be composed of four components: a radiation transport code, a depletion code, a thermal hydraulics solver, and a structural simulation code. The radiation transport code is Shift, a high-performance Monte Carlo code developed at ORNL, which will calculate heat generation in the fuel rods. The evolving nuclide inventory will be calculated by ORIGEN, a nuclide burnup and decay solver in SCALE. The thermohydraulic solver is COBRA-SFS Cycle 4a. COBRA-SFS solves subchannel equations to compute the pressure, density, and temperature in the system. Finally, the structural code to be used is Diablo, a three-dimensional structural-thermal-mechanics code. Diablo will be used to calculate the stress, strain, and deformation on the cask vessel and structural internals.

Initial development is focused on coupling Shift and COBRA-SFS for steady-state coupled transport-thermohydraulic simulations. A series of challenge problems is being developed, beginning with steady-state coupled transport-thermal hydraulics simulations of 3×3 arrays of pins, eventually proceeding up to time-dependent coupled simulations of fully loaded DPCs. This paper demonstrates the results of the initial efforts in steady-state coupled transport and thermal hydraulics applied to a 3×3 array of pins in a flooded spent-fuel canister.

14h00 - 15h40 > Track 1

SIMULATE5 ANALYSIS OF A SPENT FUEL POOL

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This paper describes proof-of-principle fuel pool criticality calculations using the nodal diffusion code SIMULATE5, which is supplied with cross section data from CASMO5. The accuracy of SIMULATE5 is tested against CASMO5 and MCNP6.2 for several demonstration problems using PWR fuel and a mock rack geometry. These test problems include various features, such as fuel with different exposures, empty locations, and a water reflector. Most of the tests are in two dimensions, but one three-dimensional test is included. For all test problems, SIMULATE5 predicts criticality within 250 pcm of the reference. Without its submesh model, these results are much worse for several of the cases (up to 2200 pcm error). As expected, CASMO5 and MCNP6.2 agree closely. For these examples, SIMULATE5 is 50x to 150x faster than CASMO5 and 10^3 to 10^5 times faster than MCNP6.2. Given its accuracy for these demonstration problems, along with its computational efficiency and ease-of-use, SIMULATE5 could be a useful tool for fuel pool criticality analysis.

TRIPOLI-4®: OVERVIEW OF THE CODE CAPABILITIES FOR CRITICALITY IN VERSION 11

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In this paper we describe the main capabilities of the continuous-energy Monte Carlo transport code TRIPOLI-4® in the field of criticality. The main focus of this work concerns the new features available in version 11, which was released in November 2018: adjoint flux calculations, fission matrices, reactivity perturbations and sensitivity of the k-eigenvalue to nuclear data, critical parameter search, and transport in stochastic geometries.

RECENT DEVELOPMENTS TO THE MONK MONTE CARLO CODE FOR CRITICALITY SAFETY AND REACTOR PHYSICS ANALYSES

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Recent developments to the MONK® Monte Carlo neutronics code and its associated nuclear data libraries and the Visual Workshop integrated development environment are described. Improved physical modelling of bound thermal scattering and scattering at epithermal resonances, together with new nuclear data libraries containing low temperature data, improve the accuracy with which the effects of temperature on criticality can be assessed. Enhanced tallying capabilities include the implementation of a continuous energy adjoint flux estimator, thermal power flux normalization and power distribution output, and a new solid sphere tally body. The MONK validation database has been improved with the addition of a significant number of Tier 2 benchmarks and validation at elevated temperatures. Optimization, uncertainty quantification and validation tools in Visual Workshop are also described.

EVALUATION OF MCNP'S FISSION MATRIX CAPABILITY FOR CRITICALITY CALCULATIONS

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Historically outcomes to solving criticality calculations using the fission matrix approach have been limited by both the choice of computational methods and computer memory limitations. Recently MCNP developers at Los Alamos National Laboratory have utilized sparse matrix storage techniques to create a new advanced fission matrix capability for criticality calculations in MCNP. This paper documents Sandia National Laboratories Nuclear Criticality Safety (NCS) Program's application and graded approach for evaluating a new pre-released advanced fission matrix capability.

The initial evaluation uses simple models of uranium and plutonium metal and solution systems. Next the capability was applied to more complex models of the Annual Core Research Reactor and the Critical (CX) Assembly located at Sandia National Laboratories. Finally, the advanced feature was applied to a few historical Nuclear Criticality Safety calculations from Sandia. This includes the use of a "Cube-of-Cubes" method similar to that presented in an ANS paper by Mark V. Mitchell [1], as well as models supporting reactor fuel storage configuration in water filled concrete tanks. These models are known to have difficulty with entropy convergence for criticality calculations, leading to challenges in predicting when convergence occurs in the model.

The results from the application of the fission matrix capability across simple and complex systems demonstrates the robustness, accuracy, and time savings from the accelerated convergence option when applying this capability in MCNP criticality calculations. Additionally, application of this capability simplifies and removes the guesswork for the initial source creation and parametric analyses for MCNP models.

[1] M.V. Mitchell, "Practical Application of the Single-Parameter Subcritical Mass Limit for Plutonium Meta", Data, Analysis, and Operations for Nuclear Criticality Safety-II

AUTOMATED ACCELERATION AND CONVERGENCE TESTING FOR MONTE CARLO NCS CALCULATIONS

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Monte Carlo methods have been used for over 60 years in nuclear criticality safety (NCS) calculations. Significant burdens are placed on NCS analysts to properly run the calculations: (1) The initial guess for fission sites is defined by user input ; (2) Users must ensure that sufficient neutrons/cycle are used to prevent bias ; and (3) Users must ensure that enough cycles are discarded so that k_{eff} and the fission source have converged. In practice, a short run produces plots of k_{eff} and entropy, then the number of inactive cycles is manually set in the mcnp6 input file, and a final run is made. NCS work often requires parameter studies with 100s of runs. For these studies, it is not practical to follow all of the procedures above, and conservative over-estimates are used for the inactive cycles. Recent work has addressed these burdens, providing automated acceleration of the convergence process, statistical tests for automatically determining convergence, and additional tests to assess whether a sufficient number of neutrons/cycle was used. These automated methods do not require user input and provide quantitative evidence of convergence. Testing on a wide range of problems has demonstrated that the methods are robust and reliable.

CRITICAL EXPERIMENT DESIGN USING OPTIMUS

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The design process for critical and subcritical experiments involves weighing the advantages of many geometric and material factors while also trying to emphasize specific characteristics of the experiment. For example, a critical experiment could seek to maximize the neutron energy spectrum in a specific energy regime in order to validate different regions of the material's cross section. As these experimental designs increase in dimensionality, so does the complexity of generating the critical configurations for the practitioner. There exists a need to intelligently automate the design process for such experiments.

The Optimus software package, developed at Lawrence Livermore National Laboratory (LLNL), addresses this need by taking a modular approach to generating the physics model, optimizing the design, and storing the results from the experimental design process. This type of modular approach gives the user flexibility in how the model is created, what physics software is used, and how the results are analyzed.

This work presents an overview of Optimus and how it has been successfully applied to designing two Thermal/Epithermal eXperiment (TEX) configurations using ^{239}Pu with alumina and ^{233}U with polyethylene. Then, Optimus is applied to a general ^{239}Pu two-region curve with polyethylene dilution. Optimus is shown to be up to 80% more computationally efficient than a brute force approach for this model. Optimus overall reduced the total number of simulations need to determine all the critical configurations by 50% over the brute force approach.

VALIDATION OF DEEP LEARNING METHODS FOR NUCLEAR CRITICALITY SAFETY

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A study was conducted to determine if it is feasible to train an ensemble model, comprised of several neural networks, to infer effective multiplication factor (k_{eff}) to high enough precision to forego further Monte Carlo radiation transport code calculations. This study consisted of training neural networks using MCNP output files that were applicable to a water-moderated and reflected plutonium sphere. This study shows that a neural network can be trained to accurately infer k_{eff}, and that it is feasible to build an ensemble model that can infer millions of k_{eff} values in a fraction of the time it takes a typical Monte Carlo radiation transport code to perform a similar set of calculations.

NEUTRON MULTIPLICATION IN FUEL-WATER RANDOM MEDIA

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In this work, we quantify the impact of the randomness of the traversed medium for a simple benchmark problem of neutron transport in a stochastic mixture composed of low-enrichment (3.7%) UO_2 fuel fragments dispersed in water. A tool that allows

sampling a large set of stochastic material configurations by Monte Carlo methods has been recently developed [1-3]. By using this tool, we assess the effects of the stochastic geometries as opposed to lattice dispersions of fuel in a moderator material. The findings of our preliminary investigation show that lattice models are not always conservative with respect to the reactivity.

[1] Larmier, C. et al., "Finite-size effects and percolation properties of Poisson geometries", Physical Review of Energy, 94, 012130 (2016).

[3] Marinosci A. et al., "Neutron transport in anisotropic random media" , Annals of Nuclear Energy, 118, 406-413 (2018).

Session 4 > Conference Room LOUIS ARMAND

12h00 - 12h50 > Track 7

EVALUATION OF THE IMPACT OF NEUTRON ABSORBER MATERIAL BLISTERING AND PITTING ON SPENT FUEL POOL REACTIVITY

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Neutron absorber materials (NAMs) are used in spent fuel pool (SFP) storage racks to increase storage capacity while maintaining criticality safety margins. Operating experience to date has revealed blistering and pitting in BORAL®, a commonly used NAM in SFPs in the United States and many other countries. The objective of this study is to generically evaluate the impact of blisters and pits on SFP reactivity. For broader applicability, simulations were performed for a generic NAM, which has no protective cladding so that the results can be applicable not only for BORAL®, but also for other metal matrix composite NAMs used in SFPs. Because BORAL® has the longest history, operating experience to date for BORAL® has been evaluated, and the worst cases—in terms of pit and blister sizes and locations—have been simulated. This part of the analysis demonstrated that even in the worst operating experience case, the impact on reactivity is insignificant (<100 pcm). Then, hypothetical extreme cases were evaluated to determine bounds for future operation. Analysis was performed for two fuel types used in pressurized water reactors (PWRs) to determine the impact of different fuel types. Simulations were performed at unborated conditions so that the results could be applicable for NAMs used in boiling water reactor (BWR) SFPs or other SFPs that do not contain boron. The paper presents the approach for the evaluation of the impact of pitting and blistering on SFP reactivity and the computational results based on operating experience and hypothetical extreme scenarios to determine the bounds for significant impact on reactivity.

SOME INSIGHTS IN CRITICALITY-SAFETY OF SPENT FUEL POOLS UNDER LOSS-OF-COOLING AND LOSS-OF-COOLANT ACCIDENT

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After the Fukushima Daiichi accident, some studies were needed to better assess the criticality margin of spent fuel pools under loss of cooling and loss of coolant accident scenarios.

This paper aims at exploring the subject by identifying the main parameters and conditions leading to a k-eff increase and by analyzing the results regarding the accident phenomenology based on thermal-hydraulic calculation results.

For that purpose, the textbook model used consists of a pool containing undamaged 17×17 PWR UOX fuel assemblies with 5 wt%. maximum enrichment. Two typical storage rack designs are considered for the study: a 'low density' one (with a rather large pitch and without poisoned structures) and a 'high density' one (with a rather small pitch and with borated structures).

CRITICALITY ANALYSIS OF THE NEW DN30 PACKAGE FOR THE TRANSPORT OF UF_6

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The DN30 package was developed by DAHER NUCLEAR TECHNOLOGIES GmbH (DNT) for the transport of enriched commercial grade and reprocessed UF_6 up to an enrichment of 5 %. It consists of a standard 30B cylinder and the DN30 Protective Structural Packaging (PSP) and is licensed as a type AF, IF and B(U)F package.

Due to its chemical and physical properties, enriched UF_6 presents several challenges to criticality safety that have to be taken into account for the safety assessment of the DN30 package. The assessment is based on variation calculations of criticality relevant parameters such as geometry and material composition for individual packages in isolation and arrays of packages. Single packages and infinite 3D arrays of packages were simulated to figure out most reactive arrangements and prove criticality safety.

Since UF_6 in 30B cylinders is undermoderated, the consideration of conservative amounts of moderation and the impact of their geometrical distribution on reactivity are of paramount importance. This involves the determination of all possible moderation sources, not only those resulting from inherent impurities in UF_6, but also from the interaction of the UF_6 content with in-leaking water vapor. Regarding impurities, the criticality analysis assumes up to 0.5 wt.% HF, and also takes new findings about hydrogenated uranium residues into account.

Based on these potential quantities of moderation, the criticality analysis is performed for a variety of geometrical fuel/moderator distributions by means of conservative calculation models, incorporating accident conditions of transport and effects related to the physical properties of UF_6.

Criticality safety was proven for the DN30 package even for very conservative, hypothetical assumptions that are unlikely to be encountered in actual packaging, transportation, and storage configurations. In addition to the overview of the criticality analysis for the newly licensed DN30 package, an outlook on potential design optimizations for higher uranium enrichments will also be presented.

ASSESSING THE EFFECTS OF LOW TEMPERATURES ON K-EFFECTIVE FOR AGR SPENT FUEL TRANSPORT PACKAGES

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EDF Energy transports irradiated AGR fuel elements from our power stations to other locations, both within the UK and internationally. These elements are transported inside Irradiated Fuel Transport Flasks, which provide for cooling and containment of the fuel as well as ensuring criticality safety.

The IAEA Regulations for Safe Transport of Radioactive Material (SSR-6) require that packages containing fissile material shall maintain subcriticality during normal and accident conditions, including the effects of changes in ambient temperature between –40°C and +38°C. Criticality safety assessments for AGR flask transport must therefore demonstrate that criticality safety is maintained over this temperature range.

EDF Energy uses the MONK code for calculations of k-effective. This is capable of representing the reactivity effects of material temperatures, both from density effects and changes in nuclear reaction cross-sections. Density effects can be assessed explicitly. However, currently released nuclear data libraries do not contain tabulations of nuclear data below Room temperature (20°C). Therefore, any calculations at lower temperatures require extrapolation outside the tabulated range. In addition, there is very little validation data available for k-effective calculations in this temperature range.

This paper describes the work done by EDF Energy and by Wood plc to assess temperature effects on flask k-effective at low temperatures. Nuclear data effects can be estimated by extrapolation from calculations at higher temperatures and estimates of the resulting uncertainty from this approach have been derived. In addition, direct calculations have been carried out using a development status nuclear data library, which includes extended tabulations at low temperatures. These two methods were found to give good agreement.

For AGR flasks, which are water moderated LEU systems, density effects have been found to be the dominant factor. Therefore the maximum k-effective occurs around 0°C. The total increase in k-effective over this temperature range is very small, and does not challenge the existing safety margins for flask transport.

EFFECT OF LOW TEMPERATURES ON CRITICALITY CALCULATION FOR THE TRANSPORT OF FISSILE MATERIAL

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Fissile package design has to meet requirements of IAEA (International Atomic Energy Agency) regulation n°SSR-6 [1] in order to ensure subcriticality in usual normal and accidental conditions of transport. For these packages, these requirements are to be taken into account for temperatures of the package between -40°C (233 K) and +38°C (311 K). However, criticality calculation tools presently used do not allow to perform calculations over the required range of temperature. Criticality calculations are mainly performed at Room temperature (~ +20°C) which is the temperature generally encountered in benchmark experiments used for nuclear data validation. In the past, this hypothesis has not been questioned since the neutron effective multiplication factor (Keff) decreases with temperature for temperature above +20°C and because of the models simplifications generally taken into account. However, since neutron cross-section for low temperatures are available on a theoretical basis, it is possible nowadays to explore if the assumption of using +20°C data is really valid over the whole range of temperature.

This paper shows that a decrease of temperature leads to a decrease of Keff due to the decrease of H_2O density when water is in the solid phase (ice) compared to the liquid phase on the base of two simplified cases considering water as the moderator. On the other hand, considering nuclear data, a decrease of temperature could lead to an increase of reactivity in particular due to the Doppler effect on ^{238}U capture cross section.

Nevertheless, new studies should be performed in order to better understand the global effects of low temperature on nuclear data (for example, with more temperatures and nuclides). In particular, the impact of density variation of polyethylene as moderator must be assessed as its behaviour could be very different from those of water.

[1] Regulations for the safe transport of radioactive material, SSR-6, 2018 edition, IAEA Safety Standards.

AWG-711, A TYPE C TRANSPORT PACKAGE

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To effectively and securely allow air transport of fissile material, a Type C transport package has been developed in collaboration with Sandia National Laboratories. The package has been designed to comply with the requirements set out in both the IAEA Transport Regulations and the American NRC Transport Regulations. The package design is complete and four full scale packages have been manufactured. Three of these have been used in trials to underpin the tests set out in the relevant regulations.

This package, the AWG 711, has undergone a detailed engineering design process with input from the Criticality Safety Group at AWE. This process ensures that the final design meets the stringent requirements of Type C transport package certification enabling the required fissile payload(s) to be carried. Details of the package design, trials and the supporting criticality calculations are presented.

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16h10 - 17h50 > Track 4

INVESTIGATION OF INFERRED PARAMETERS IN SUBCRITICAL EXPERIMENTS

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Subcritical measurements involve measuring the neutron and/or gamma radiation emitted from a Special Nuclear Material (SNM) system of interest. Typically these measurements involve the use of a data acquisition system that can provide temporal information. Such information is used in neutron noise measurements to infer information about the SNM system. The parameter that the criticality safety community is generally the most interested in is the multiplication factor, k_{eff}. It is important to note that many of these parameters (including k_{eff}) are inferred, not directly measured. The inference model(s) used to derive these parameters include various assumptions. This work investigates some of these assumptions and their applicability. Recent subcritical benchmarks performed at the National Criticality Experiments Research Center (NCERC) are presented. This work details some results inferred from measurements and simulations for these systems. Last, this work uses simulated data to explore the question of how multiplication and k_{eff} change for small perturbations over a large range of reactivity states, which is useful to help determine which parameters are valid over which reactivity ranges.

MEASUREMENT OF SUBCRITICALITY IN DOLLAR UNITS USING TIME-DOMAIN DECOMPOSITION BASED INTEGRAL METHOD

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This paper presents an estimation method of subcriticality in dollar units developed on the basis of an integral method for arbitrary state changes in a subcritical system. In a general transient in a subcritical system, reactivity, neutron source intensity, and point kinetics parameters (Λ and β_{eff}) can vary simultaneously. To address this problem, the "time-domain decomposition based integral method (TDDI)" has been proposed. The TDDI method can estimate the subcriticality only using the time variation of the neutron count rate. Therefore, the proposed method is useful to estimate the subcriticality in the reactor where the detailed conditions are unknown. To investigate the applicability of the TDDI method to actual subcritical measurement, a transient experiment in a source-driven subcritical system is conducted at the Kyoto University Critical Assembly. As a result, it is concluded that the TDDI method can approximately estimate the order of magnitude for the subcriticality in dollar units. Meanwhile, the estimation is difficult owing to the statistical errors when neutron count rate is low.

MUSIC: A CRITICAL AND SUBCRITICAL EXPERIMENT MEASURING HIGHLY ENRICHED URANIUM SHELLS

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In the International Handbook of Evaluated Criticality Safety Benchmark Experiments, the number of subcritical neutron multiplication benchmarks is still quite small relative to the number of critical benchmarks. These subcritical benchmarks are important, as some aspects of nuclear data such as the probabilistic distributions of neutrons emitted in fission are used in subcritical multiplication measurements, but not in critical experiments. Critical experiments are more sensitive to average values. To expand this range of subcritical benchmarks, an experiment will be performed in collaboration with IRSN that features highly enriched uranium shells, known as the Rocky Flats shells, measured by a variety of neutron multiplicity detectors. The goal is for this experiment to aid in the validation of subcritical measurement and simulation methods, along with nuclear data. This experiment would also build upon the lessons learned in previous subcritical experiments performed at the National Criticality Experiments Research Center (NCERC).

This experiment will be performed using the Planet vertical lift machine, which will combine an upper and lower subassembly together to assemble the full experimental configurations. These configurations will consist of varying amounts of the Rocky Flats shells to achieve neutron multiplications from deeply subcritical (effective multiplication factor κ_{eff} of ~0.649) to critical. Subcritical configurations would be measured by multiple neutron detectors, including the Neutron Multiplicity Array Detector (NoMAD) that has been previously used for a number of different NCERC subcritical experiments. Additional neutron sources such as californium or a D-T neutron generator will be also be used for some measurements.

The experiment proposed here will help improve nuclear data for uranium, and validate both subcritical simulation and measurement methods in a way that the numerous uranium critical benchmarks cannot. With the eventual goal of inclusion into the ICSBEP handbook, this would be the benchmark with the largest range of multiplication values, enhancing its usefulness as a validation tool.

Session 1 > Room A-B

9h00 - 10h40 > Track 6

THE DOUBLE CONTROL AND ITS CONSISTENCY WITH THE DOUBLE CONTINGENCY PRINCIPLE

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The double contingency principle is implemented in French nuclear research and development facilities. Regarding nuclear criticality safety in the CEA's Cadarache centre, the double control principle was initiated in the LECA-STAR facility in the early 2000's and generalized to the other (approx. 20) nuclear installations since then.

This paper presents the operational experience gathered from the beginning until now. Some simple statistics are used to illustrate the reliability of the double control principle as applied in the CEA/Cadarache facilities. The significant events registered at the ASN (Autorité de Sûreté Nucléaire – Nuclear Safety Authority) are analysed to demonstrate that defence in depth prevails and the failures that have occurred were detected and corrected by appropriate measures.

DEVELOPMENT OF A UK WORKING PARTY ON CRITICALITY LEARNING FROM EXPERIENCE DATABASE

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The UK's Working Party on Criticality (WPC) is the key national committee focused on criticality safety issues, with one of its key objectives being "to provide a forum for the discussion and distribution of information of relevance to criticality safety in the UK". Three years ago, the WPC members highlighted that a key area where value could be added for industry was Learning from Experience (LFE). In response, a WPC sub-group was set-up to focus on this topic; the progress of this sub-group forms the subject of this paper. LFE is considered essential in maintaining and improving safety and preventing accidents across the industry. Indeed, the UK's regulator, the Office for Nuclear Regulation (ONR) highlights the importance of organisations learning from internal and external sources in its own regulatory guidance.

Historically LFE data has been shared through the WPC in the form of entries in WPC member reports; presentations at WPC meetings covering national and international near misses or accidents; informal discussions at meetings and workshops and international briefing notes distributed via email. However, the group noted that:

- Although a good amount of high quality LFE information is and has been disseminated through the WPC from both national and international sources, this was not collated in a format where it would be easy to access in the future.
- High level events (which attract regulator scrutiny) are generally well shared at the WPC. However, the group noted that each site is likely to have a much greater number of lower level events that are not so readily shared but yet may provide insight into common problems, ways to identify them and potential solutions. This would be valuable for other members of the UK's criticality safety community performing similar operations.
- In a commercial world, many organisations have reputational sensitivities that can make sharing LFE with the wider community challenging.
- An overview of the types of incidents happening in members organisations could indicate trends or common problems that the WPC would be well placed to help address via discussion or the work of a sub-group for example to the benefit of all.

The group therefore determined that a good solution would be the creation of a WPC LFE database which would be stored securely and populated anonymously. Populating this database with historic events would be problematic and resource intensive; however, it is noted that entering in events as they occur would be a much more manageable endeavour and lead over time to an invaluable resource for the community. A WPC LFE coordinator role has been established to encourage its population and provide feedback to the members. This paper provides more detail on the database and how it has progressed to date.

NUCLEAR CRITICALITY SAFETY LESSONS LEARNED IN THE DESIGN OF THE URANIUM PROCESSING FACILITY AT THE Y-12 NATIONAL SECURITY COMPLEX

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The Uranium Processing Facility (UPF) is being built by the National Nuclear Security Administration (NNSA) to replace aging uranium processes at the Y-12 National Security Complex (Y-12 NSC) in Oak Ridge Tennessee. Specifically, the processes housed in building 9212 are being replaced as this facility dates to the World War II Manhattan Project days of Y-12. When the UPF Project is completed it will consist of three main facilities along with several support facilities. The Main Processing Building (MPB) will house oxide production and metal casting operations. The Salvage and Accountability Building (SAB) will house wet chemistry operations (examples include recovery evaporation and calcination of solutions). The Personnel Support Building will be connected to both the SAB and the MPB and will serve as a transition facility for materials and personnel between the two. The MPB will also be connected to the Highly Enriched Uranium Materials Storage Facility (HEUMF) via a connector (HCON) to permit transfers from storage into UPF directly.

This paper will summarize some of the more important lessons learned for the NCS practitioner working in a design environment rather than in a production or experimental environment. As the Chief NCS Engineer for the UPF Project I have been responsible for ensuring the technical evaluations produced on the project are high quality, technically accurate, and conform to the Y-12 NCS Program expectations. Ensuring nuclear criticality safety for processes that do not yet exist in reality is a challenge very different than that of ensuring subcriticality for processes in an operating facility. The biggest difference being that in design space – everything is imaginary whereas in the operating facility you have the benefit of being able to go out and look at the equipment and interview the operators. Ensuring that Nuclear Criticality Safety is integrated into design is a process that requires constant communication between the NCS staff and the broader engineering design team. Until the facility is constructed – nothing is physically stable and so rigorous documentation of decisions – to include any assumptions – is necessary to minimize errors. Non-NCS engineers will not understand the hazard or the physics of nuclear criticality. It is up to the NCS staff to provide training and communication to the team. The NCS staff must be patient and prepared to explain themselves many times and be as flexible as possible.

11h10 - 12h50 > Track 6

THE USE OF A HAND-HELD ENRICHMENT DEVICE IN SUPPORT OF URANIUM RESIDUE RECOVERY – A BENEFIT OR FALSE CONFIDENCE?

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The National Nuclear Laboratory's Preston Laboratory in the United Kingdom (UK) was constructed in the 1990s and is designed to service the needs of businesses with regards to low activity uranium research and development. One of the key areas in the laboratory is the Pilot Plant which is dedicated to the clean-up of residue arisings (both current and legacy) from various UK and international sites, with materials up to 5.0 $^{w}/_{o}$ ^{235}U enrichment routinely processed. From the criticality safety perspective, the facility consists of a number of large tanks / vessels and hence criticality control is primarily based on limiting the fissile mass in a process "bay". This is fulfilled through a series of administrative measures that, amongst other things, determines the level of confidence in the assigned enrichment (and hence the degree of confirmatory analysis and / or conservatism required).

During a long term safety case review of the facility, the strong reliance on administrative measures was identified and hence efforts were directed at potential means of reducing the vulnerability to operator error. One of the options investigated was the potential benefit of utilizing a hand-held enrichment device to provide an independent means of confirming the assigned enrichment. Following a detailed review of the Low Resolution Gamma Spectroscopy technique employed by the device (including limitations), a small plant trial was undertaken to determine if the device could be used more frequently and reliably as a rapid initial assessment of the enrichment. As expected from the theory, the trial confirmed that the device would be suitable for the following three general aspects:

- Roughly distinguishing between depleted, natural, enriched and > 5.0 $^{w}/_{o}$ ^{235}U enriched materials, with the accuracy of the enrichment reading increasing markedly for higher uranium concentration materials;
- Providing a rough indication of the uranium content of containers, based on the associated count rate (i.e. an indication of high or low uranic inventory);
- Detecting inhomogeneity in containers, both in enrichment and uranium content.

Based on these observations, recommendations were made relating to the future advised use of a hand-held enrichment device as a confirmatory enrichment check. A key point influencing these recommendations was the compelling case that routinely using a device which is actually expected to give imprecise / inaccurate enrichment results for low uranium content residues (characteristic of most material types received into the Pilot Plant) may actually compromise the integrity and value of the check – in essence, providing false confidence if not used selectively in a smart fashion. The importance of the readings from the device being carefully considered by Suitably Qualified and Experienced Persons who understand the physics (and hence the limitations of the technique) should also not be underestimated.

In conclusion, the two situations where the use of a hand-held enrichment device has been deemed to be of most potential benefit are where the expected material type has a high uranium content , or there is some degree of doubt / concern about the assigned container inventory. The use of the device is now routinely considered as part of the criticality control regime for the Pilot Plant, although recent improvements in the local analysis capability have enabled an increase in confirmatory sampling and hence lessened the need for the device.

NUCLEAR CRITICALITY SAFETY ANALYSIS: RECOVERY OF OLD CONTAINERS HOLDING FISSILE MATERIAL

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This paper presents a nuclear criticality safety analysis of an operation to recover containers from a 30-year-old storage site. There are 145 containers containing waste from fuel elements with fissile material. The storage consists of 15 cement wells, 7 m deep and 0.4 m in diametre. Each well contains up to 10 60-liter containers stacked on top of each other.

Waste containers contain mixtures of various fuel element sections and samples which have various shapes (rods, needles, bars, etc.), different physical or chemical forms (metal, oxide) and variable isotopic compositions. The analysis is based on an inventory of the quantities of fissile material and their composition, which are well known.

Given the duration of storage and the possibility of dropping a container during handling, two methods of taking up containers are envisaged:

- handling the container as a whole (without opening the container),
- opening the container inside the well and recovering the waste directly into the container.

To choose between the two methods, it is necessary to ensure that the grouping of fissile material present in two stacked containers is lower than the safe mass limits for the fissile reference medium of these two containers.

The paper presents the method for determining a reference fissile medium that bounds all fissile material present in the two containers without being too penalizing. It takes into account the nature of the fissile material (oxide or metallic, homogeneous or heterogeneous), ^{235}U enrichment and the moderating material. The paper also presents calculations of critical and safe mass limits of fissile material for all the reference fissile media.

For each grouping of two containers, the paper presents the calculations of an equivalent mass of fissile material in both containers and then verifies that it is below the safe mass limit.

Thanks to this method, it is possible to handle 130 containers as a whole (without opening) thus saving time and reducing worker exposure for the operation all the while ensuring nuclear criticality safety control.

IMPROVED SAFETY BASIS FOR LIQUID WASTE PROCESSING AT BWXT

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The operational cost of an overly conservative safety basis can be significant. BWX Technologies, Inc. (BWXT) recently upgraded the safety basis for the liquid waste treatment facility at its Lynchburg, Virginia manufacturing plant to remove excess conservatism and allow more efficient operation. The previous safety basis used a simple water-reflected homogeneous sphere of fully enriched uranium and water without crediting the design of the equipment. This approach resulted in a maximum mass in the treatment process at 600 grams ^{235}U.

The revised safety basis being developed uses the nature of the process and the design of the equipment to establish limits. The analysis considers precipitation and extraction by organics as an upset condition in the start of the process and as a normal part of the process in the later steps. Engineers evaluated each of the tanks were evaluated over a range of sediment densities and distributions. The analyses of individual tanks established the actual margin of safety in the process. The solidification and drying equipment were likewise evaluated.

Comparing the estimated critical mass values with the previous approach demonstrates the excess conservatism that was present.

PREVIOUS BASIS:

- Critical Mass: 820 gram ^{235}U

REVISED BASIS:

- Equipment: Critical Mass
- Equalization Tanks: 2200 grams ^{235}U
- Vertical Tanks: 2800 grams ^{235}U
- Filter Press: 3200 grams ^{235}U
- Hopper: 2400 grams ^{235}U
- Drums: 2000 grams ^{235}U

The revised safety basis has established mass limits larger than can be used based on other regulatory considerations. Discussions with the United States Nuclear Regulatory Commission (USNRC) are on-going regarding revision to existing license conditions which would allow fully utilizing the increased limits in the revised safety basis.

DEVELOPMENT OF A CRITICALITY SAFETY CASE FOR WASTE RETRIEVAL FROM A HISTORICAL WASTE STORAGE FACILITY

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A historical Magnox fuel element debris storage facility presents one of the highest radiological hazards on the Sellafield Site. Preparations to retrieve the solid radiological inventory into waste packages and put these into modern storage facilities are reaching fruition, with retrievals due to commence later this year.

The vast majority of the Magnox fuel fissile waste in the facility presents no credible criticality risk. A very small mass fraction of the waste, Enriched Fissile Tippings (EFT), presents a theoretical criticality risk during the operations to retrieve and safe-store the waste. At the onset of work on the criticality safety case the understanding of the EFT inventory was inadequate and it appeared that the criticality risk could be near to a level regarded as intolerable in UK custom and practice.

The criticality safety case has required a detailed investigation of the records concerning the EFT. This has been a challenging task requiring expert identification, interpretation and reconciliation of records from multiple archives and document stores. As this work has progressed, our understanding of the uncertainties in the data and our level of confidence in the data has improved, to the extent that it is now considered adequate to underpin the criticality safety case.

Various theoretical accumulations and arrangements of the EFT have been modelled and used to map a 'criticality safety envelope'. The developed EFT inventory (with uncertainties) has then been compared against the safety envelope and it has been demonstrated that the likelihood of sufficient EFT accumulating in an arrangement that could cause criticality is very low.

The criticality risk is now demonstrated to be tolerably low within the context of the overwhelming need to reduce the high radiological risk as soon as reasonably practicable. Furthermore, the susceptibility of criticality to uncertainties and unknown factors is considered so low that there would be no benefit from having a Criticality Emergency Plan (CEP) and hence no requirement for a Criticality Warning System.

The paper will discuss the investigation of the EFT inventory, the development of the criticality safety case and the consideration of the need for a CEP.

14h00 - 16h05 > Track 5

PERIODIC SAFETY REVIEW IN FRANCE – FOCUS ON NUCLEAR CRITICALITY SAFETY

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This article presents the periodic safety review performed for French nuclear facilities with a focus on Nuclear Criticality Safety (NCS). Among the requirements a licensee must comply with for his nuclear facilities, the articles L. 593-18 and L 593-19 of the environmental code states that a licensee must carry out a periodic safety review (PSR).

It says that "the licensee of a nuclear installation carries out a safety reassessment of its facility periodically, taking into account the best international practices. This review must allow a clear view of the facility's situation in regard to the applicable rules, and update the risks or drawbacks concerning the facility (...) taking into account in particular the facility's state, its operating feedback, the evolution of knowledge and of rules applied to similar facilities. The periodic safety review should occur every ten years. However, the creation decree could define a different periodicity depending on the facility's specific characteristics. The licensee must send to the Nuclear Safety Authority and to the ministries in charge of nuclear safety a report setting out the reassessment conclusions and, if necessary, the measures to be taken in order to mitigate any observed non-conformity or to improve safety."

The article will present the objectives of the PSR, the expected content of the PSR report written by the licensee, a detail description of the NCS issues related to the PSR (with several examples) and the schedule and proceedings of its review.

Finally, this article will show that the PSR is a major step of continuous nuclear safety improvement in France and a good opportunity to improve transparency, especially in the context of life extension of nuclear facilities.

DEVELOPMENT OF AN ISO STANDARD RELATED TO GEOMETRICAL DIMENSIONS FOR SUBCRITICALITY CONTROL

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Nuclear criticality safety is often based on requirements limiting geometrical dimensions. This is usually expressed as maximal dimensions for process equipment containing fissile material (e.g. maximal diameter of columns, maximal height of a stack...) or minimal dimensions for interactions (e.g. storage pitch) or neutron absorber dimensions (e.g. thickness of a cadmium sheet). These limits are derived from nuclear criticality safety calculations and assessment.

The nuclear criticality safety specialist needs to deal with different topics: how to avoid an intractable amount of limits while maintaining an adequate safety level, how to comply with the requirements, what should be compared.

In this context, an ISO (International Organization for Standardization) standard has been developed (ISO 21391) by Subcommittee SC 5 (Nuclear installations, processes and technologies) of Technical Committee ISO/TC 85 (Nuclear energy, nuclear technologies, and radiological protection), in order to provide guidance, requirements and recommendations related to the determination of the relevant dimensions to be ensured on the one hand, and to the comparison between actual dimensions and subcriticality dimensions on the other hand.

This paper presents the main items, with examples illustrating how to avoid unnecessary complexifications, why some dimensions are better suited for controls, how to deal with aging effects and deformations due to aggressions to be taken into account. These examples are analyzed either from the point of view of the design engineer or of the operation team.

It also briefly touches which topics are commonly linked to the geometrical dimensions (e.g. neutron absorber content).

REPROCESSING FACILITY PERIODIC SAFETY REVIEW: HOW IMPACT OF AGING EFFECTS ON GEOMETRICALLY SAFE EQUIPMENT IS REVIEWED

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In France, according to the law, all nuclear fuel cycle facilities have to undergo a Periodic Safety Review every ten years. Concerning UP2-800, a spent fuel reprocessing facility of La Hague nuclear site, one point specially reviewed is the impact of aging effects on the Nuclear Criticality Safety of this facility. Indeed, dimensions of geometrically safe equipment and chemical compositions of neutron absorbers can be affected by aging effects. This paper presents a methodology used for reviewing these impacts and some results of this review transmitted to the French Nuclear Safety Authority.

CLAIMS-ARGUMENTS-EVIDENCE

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Within the field of criticality safety there is often a need to present technical information to a range of stakeholders including operators, senior management and regulators. When communicating this information there is a need to ensure that the information provided is clearly presented, logical and robustly underpinned.

At Sellafield Ltd a logical thinking tool called Claims Arguments Evidence (CAE) is in widespread use. CAE is a simple yet effective way to develop and present a technical argument that provides a logical structure; setting out the claims required to be made and the arguments and evidence that support them.

The benefits of this approach include providing clarity on what is being claimed, supporting effective safety case implementation (by ensuring that there is a clear link between safety claims and safety designations), being able to readily identify important gaps in knowledge and initiate activities to address these, and enabling improved stakeholder engagement.

The use of CAE within safety assessments, technical reports and wider safety case documentation is now common place at Sellafield Ltd, where it has been used to support the diverse array of operations undertaken on the Sellafield site ranging from small scale analytical operations to large scale projects. Safety case documentation built upon the CAE logic has been given extremely positive feedback from Sellafield Ltd Nuclear Safety Committees and the UK regulator, the Office for Nuclear Regulation (ONR).

This paper will explore what CAE is in greater depth, discussing the terminology and the guidance that has been developed, explore examples of its use at Sellafield Ltd, and highlight the benefits this has provided.

USE OF BARRIER ASSESSMENT IN CRITICALITY FAULT ANALYSIS

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In common with much of UK practice, criticality fault analysis at Sellafield Ltd has for the last 20 years primarily utilised the Design Basis Analysis Assessment (DBAA) tool. DBAA assigns a pre-defined level of defence in depth based on consequence and likelihood. Typically for a criticality fault this requires the designation of two lines of defence. The lines of defence are required to be robust and fully independent from each other and the way the fault begins (known as the initiating event).

The application of DBAA for criticality fault analysis has proven problematic on numerous occasions. For example, where there is difficulty in defining a meaningful initiating event or on older facilities where the process was not designed with DBAA in mind and involves complex operational procedures and maintenance. This has led to over-designation and excessive effort being spent on perceived shortfalls in the provision of safety measures.

Therefore, other tools for carrying out criticality fault analysis have been explored and developed. Barrier assessment is one of these tools. It lays out all of the barriers which prevent a potential fault resulting in a consequence. It is particularly effective at giving a greater understanding of various factors that contribute to overall criticality risk, especially if there is a large reliance on operational controls or if the controls are not fully independent.

The use of barrier assessment was accepted in principle by both the UK Working Party of Criticality (WPC) and the UK regulator. the Office for Nuclear Regulation (ONR). More importantly the technique has been commended by those responsible for maintaining safety at Sellafield Ltd facilities. The paper will provide details of how to use the barrier assessment technique including real examples from Sellafield Ltd.

Session 2 > ❷ Room C-D

9h00 - 10h40 > Track 3

EFFECT AND UNCERTAINTIES OF H IN ICE THERMAL SCATTERING LAWS ON THE NEUTRON MULTIPLICATION FACTOR FOR PWR FUEL CRITICALITY APPLICATIONS

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In the context of the IAEA recommendations to ensure the transportation of fuel assemblies between 233K and 311K, thermal scattering laws of hydrogen in iced water have been produced with the LEAPR module of the NJOY code and included in the JEFF-3.3 nuclear data evaluation. Following this work, a benchmark was launched by the OECD/NEA Working Party on Nuclear Criticality-Safety subgroup-3 to evaluate the effect of the temperature on the criticality of a PWR assembly. This paper first focuses on the results obtained by CEA with the TRIPOLI-4® Monte-Carlo code on this benchmark. They show that computations made at 293K are conservative -in terms of criticality-safety- and that the density impact on the k_{eff} is much stronger than the nature of the hydrogen bound or the adjustment of nuclear data to temperature. To go further, the uncertainty of the thermal scattering laws of hydrogen in iced water have been evaluated and propagated on one of the benchmark cases. To compare their relevance, two methods were used to do so, one consisting in a direct propagation of the LEAPR model parameters uncertainties and the other one using covariance matrix of the hydrogen in iced water scattering cross section. The direct propagation, considered as the reference method here, leads to an uncertainty of 111pcm. The uncertainty evaluated with the second method is lower by around 50pcm. Whatever the method considered, those uncertainties remain low in the criticality-safety context especially as the effect of the temperature on the k_{eff} and the impact of the hydrogen bound nature are both low regarding density effects.

REPRESENTATIVITY ANALYSIS IN REACTOR CORE CALCULATIONS.

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Representativity quantifies the similarity between two cases in terms of uncertainties for a given quantity of interest. It's determined through the calculation of the sensitivities of a parameter to nuclear data for the two different cases combined with the associated covariance data.

Reactor operators have experimental data based on physical tests done during startup phases. This physical tests allows for the qualification of calculations of quantities of interest for safety, in particular criticality safety, such as critical boron concentrations, control rod worth or temperature coefficients. Those safety related quantities evolves with burn ups but are not measured on line. In reactor physics safety analysis, static full core calculations are done with few group cross sections based on data previously calculated at assembly level with evolving burn ups. To progress toward the calculation of representativity

of measured situations with the one of potential situations of accidents, this paper presents some intermediate representativity calculations. Our evaluations of the representativeness of smaller scale calculations when compared to larger scale, as well as the correlations between the situations occurring with different fuel burn-up are expected to be useful for the safety demonstration. In the future, this analysis could be performed for other quantities of interest.

SENSITIVITY AND UNCERTAINTY BASED TECHNIQUES TO EXTEND THE DATABASE OF EXPERIMENTAL VALIDATION BENCHMARKS: PRACTICAL EXAMPLE OF USE FOR TRIGA FUEL

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Analysis of sensitivity profiles and covariance data provides a robust mathematical approach towards assessing the similarity of experimental benchmarks that can be used for validation of criticality calculations. This approach is all the more valuable when dealing with fissile materials for which only a small number of experimental benchmarks exist, since it makes it possible to select other relevant experiments performed with different fissile materials.

A practical example of Framatome industrial application is performed on TRIGA fuel, used worldwide in research nuclear reactors. The fissile material of such fuel elements is a metallic alloy of UZrHx intrinsically moderated with an atomic ratio of H/Zr close to 1.6 and a U content ranging between 8 and 47 wt.%, with 20 wt.% enrichment.

The experimental validation of such media is typically based on a very small number of ICSBEP (International Criticality Safety Benchmark Evaluation Project) experimental configurations (mainly IEU-COMP-THERM-003). However, the analysis of sensitivity profiles, as well as the contribution of each reaction to prior uncertainty using various covariance data helps pointing out other ICSBEP benchmarks of interest. These experiments can then be used to extend the experimental validation database and justify the bias due to nuclear data and calculation scheme for criticality calculations performed with the French criticality safety code package CRISTAL.

Those calculations, currently performed with the TSUNAMI/TSURFER sensitivity/uncertainty sequence of the SCALE package, show the potential industrial interest in these approaches. They also highlight the need to develop in-house expertise in the domain, as well as tools more consistent with CRISTAL package such as MACSENS, currently under development at IRSN.

WHISPER S/U BENCHMARK ANALYSIS OF METAL-WATER CRITICAL MASS CURVES

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The purpose of this paper is to discuss the application of Sandia National Laboratories' (Sandia's) generic standard set of ICSBEP benchmarks, the set of benchmarks shipped with the Whisper-1.1 software (LANL), and the combined set of benchmarks in support of criticality calculations with MCNP-6.2. Sensitivity and uncertainty (S/U) analysis was performed with Whisper-1.1 software to determine similarity between the benchmarks and the computational models of spherical fissile metal-water mixtures that are formulated to compose the critical mass curves of ^{235}U and ^{239}Pu [1]. Relationships between the chosen benchmark set and the observed results from the Whisper S/U software are discussed, including discussions of the correlation coefficient, c_k, software weighting factors, and the upper subcritical limit (USL) of the effective neutron multiplication factor, k_{eff}.

The results indicate that all studied benchmark suites are sufficient to support reasonable criticality limits along the critical mass curves for ^{235}U and ^{239}Pu. The Sandia benchmarks are strongly correlated to the models that compose the critical mass curves in most of the thermal range and near the fast/metal region for both nuclides. As expected, the number of benchmarks with very high similarity decreases near the minimum concentration necessary to achieve criticality and in the very poorly moderated region. This trend is shown for ^{235}U in *(Figures 1-2)*, with the contribution from various benchmark system types detailed.

To improve similarity in regions that have lower values of c_k (e.g., <0.9), a greater number of applicable benchmarks is required. *(Figure 6)* shows the average c_k of Whisper-1.1 benchmarks when using the generic set of Sandia ICSBEP benchmarks, the ICSBEP benchmarks shipped with Whisper-1.1 (LANL), and a combined suite of benchmarks from both sources, with and without specific benchmarks that fail statistical tests (e.g., chi-squared) excluded for both ^{235}U and ^{239}Pu curves. *(Figure 3)* shows the incremental improvements in average c_k that occur when benchmark suites are combined. These improvements are most noticeable in regions with lower values of c_k.

While c_k improves with the addition of more benchmarks, the Whisper-calculated USL value does not necessarily improve, as shown in *(Figure 4)* for ^{235}U and *(Figure 5)* for ^{239}Pu. In addition to providing the Whisper-calculated USL for different benchmark suites, *(Figure 5)* highlights which individual benchmarks have the largest effect on the calculated baseline USL. A noticeable increase in USL is observed when these highlighted benchmarks are removed from the suite.

[1] Regulations for the safe transport of radioactive material, SSR-6, 2018 edition, IAEA Safety Standards.

Figure 1. Sandia benchmark suite applicability for the ^{235}U Critical Mass Curve

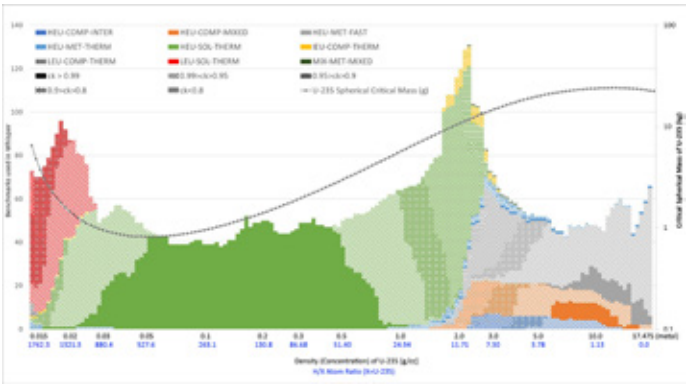


Figure 2. Combined Sandia/LANL benchmark suite applicability for the ^{235}U Critical Mass Curve

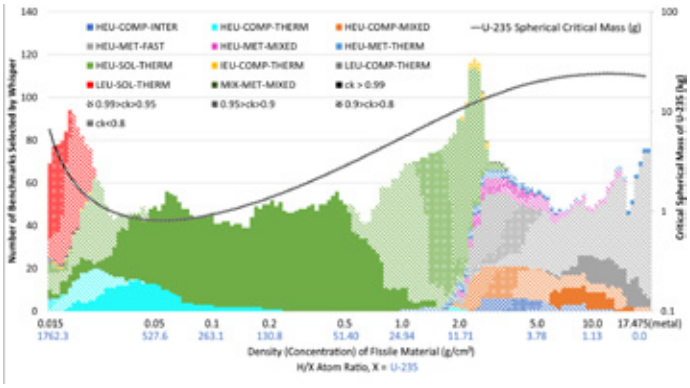


Figure 3. Average ck value for ^{235}U and ^{239}Pu critical mass curves for each analyzed benchmark suite

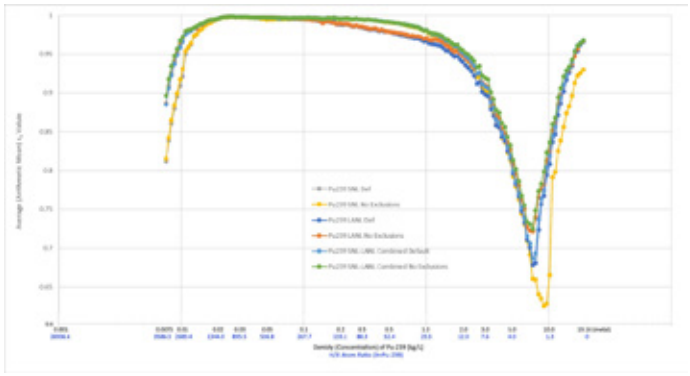


Figure 4. Whisper baseline USL of keff for the ^{235}U critical mass curve using different benchmark suites

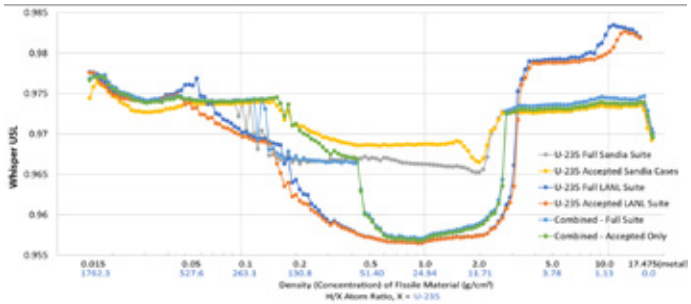
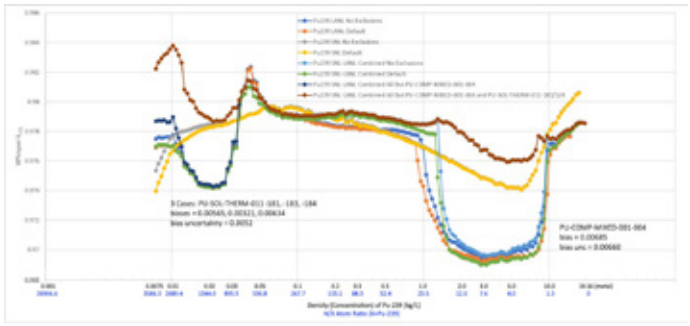

Whisper USL
U-235 Full Sandia Suite
U-235 Accepted Sandia Cases
U-235 Full LANL Suite
U-235 Accepted LANL Suite
Combined - Full Suite
Combined - Accepted Only
Density (Concentration) of Fissile Material (g/cm³)
H/X Atom Ratio, X = U-235

Figure 5. Whisper baseline USL of keff for the ^{239}Pu critical mass curve using different benchmark suites, showing the effect of including/removing high-bias benchmarks



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This paper describes objective technical results and analysis. Any subjective views or opinions that might be expressed in the paper do not necessarily represent the views of the U.S. Department of Energy or the United States Government.

11h10 - 12h50 > Track 3

USE OF WHISPER S/U TECHNIQUES IN SUPPORT OF BENCHMARK IDENTIFICATION

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The purpose of this paper is to document the first use of sensitivity and uncertainty (S/U) techniques in support of Nuclear Criticality Safety (NCS) evaluations at Sandia National Laboratories (Sandia). This paper provides the results from a S/U analysis using a pre-released version of Whisper (1.0). It will discuss how the results are similar to, and in some cases differed from the current reliance on expert professional judgment. This technique was applied to a unique application associated with three Sandia Degraded Core Coolability (DCC) experimental assemblies that have a fuel matrix of uranium oxide mixed with gadolinium oxide (UO_2/Gd_2O_3). The DCC's were

originally constructed in the 1980's and contain a homogenous debris bed fuel matrix with 24 to 27 kg UO_2/Gd_2O_3, at 11 weight percent enriched uranium and contain between 1 and 5 atom percent Gd. The fuel matrix is composed of small and medium sized particles submerged in a water bath. Each DCC assembly was irradiated in a reactor then placed in storage.

PARAMETRIC ANALYSIS OF HANDBOOK METAL-WATER CRITICAL MASS CURVES WITH MCNP

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The purpose of this paper is to present and discuss the methodology, results, and conclusions of an analysis of spherical, homogenous metal-water mixtures in MCNP-6.2 with ENDF/B-VII.1 data. The parametric analysis reproduced curves of critical fissile mass as a function of metal-water mixture density (i.e., concentration, H/X) that have been published in criticality safety handbooks [1-5]. Curves are provided for highly-enriched uranium (HEU) and plutonium-239. Furthermore, the sensitivity of the effective neutron multiplication factor to changes in mass or cross-section data is plotted and discussed.

The results from MCNP-6.2 support the existing handbook curves relating critical mass to fissile concentration for metal-water mixtures. This behavior is shown in Figures 1-2 for ^{235}U and ^{239}Pu, respectively. From these plots, it is shown that MCNP-6.2 with ENDF/B-VII.1 data generally agree with the handbook curves [1,2,4,5] for ^{235}U to within +/- 5%, with certain regions differing by up to approximately +/- 10%. However, MCNP-6.2 does not agree as well with the curves for ^{239}Pu, with MCNP generally predicting smaller critical mass values than the handbook curves from [1,3] (there is good agreement with the curve from [2], which uses MCNP-4A calculations with ENDF/B-V nuclear data and was published in 1996).

The incremental reactivity worth of incremental additions of HEU or ^{239}Pu is provided with error bars in Figure 3. This plot can be used to relate margin in USL to margin in mass limits at various concentrations. For example, at 0.1 g/cc each additional gram of HEU or ^{239}Pu increases k_{eff} by approximately 0.028 and, a USL of 0.95 would imply a margin of approximately 175 g relative to a k_{eff} of 1.0. Figure 3 also shows that incremental mass changes in ^{239}Pu are more reactive than HEU at very high (metal) and very low (<0.3g/cm^3) concentrations near criticality.

Curves calculated for subcritical limits are also presented in Figure 4 and discussed. These subcritical curves have a similar shape to the critical curves presented in Figures 1-2 but have critical mass values reduced by a greater percentage than the percent reduction in multiplication factor (e.g., a reduction in k_{eff} from 1.00 to 0.95 results in a critical mass decrease of much greater than 5%).

Lastly, the sensitivity of k_{eff} to perturbations in fission cross-section data is quantified and plotted for HEU and ^{239}Pu systems in Figures 5-6. These plots help to characterize the systems, provide a basis for comparison to other applications, and identify regions of increased importance in the underlying nuclear data.

[1] H. C. Paxton, N. L. Pruvost, LA-10860-MS: Critical Dimensions of Systems Containing 235U, 239Pu, and 233U, Los Alamos National Laboratory (1987).

[2] N. L. Pruvost, H. C. Paxton (Ed.), LA-12808: Nuclear Criticality Safety Guide, Los Alamos National Laboratory (1996).

[3] R. D. Carter, G. R. Kiel, K. R. Ridgway, ARH-600: Criticality Handbook Volume II, Atlantic Richfield Hanford Company (1969).

[4] J. T. Thomas (Ed.), TID-7016 Rev. 2: Nuclear Safety Guide (NUREG/CR-0095), Union Carbide Corporation (1978).

[5] C. B. Mills, LA-3219-MS: Critical Assemblies of Fissionable Materials, Los Alamos Scientific Laboratory of the University of California (1959).

Figure 1. Comparison of MCNP-calculated ^{235}U critical mass with handbook correlations as a function of ^{235}U concentration

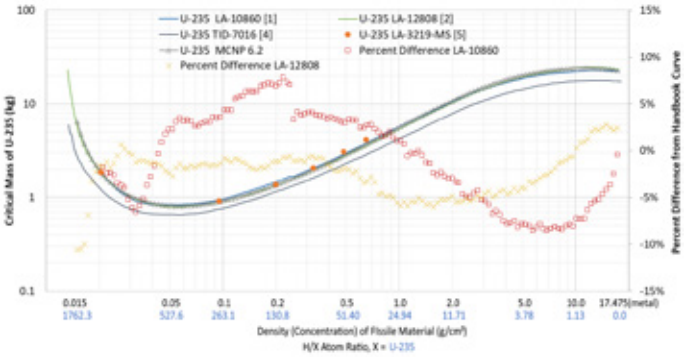


Figure 2. Comparison of MCNP-calculated 239Pu critical mass with handbook correlations as a function of 239Pu concentration

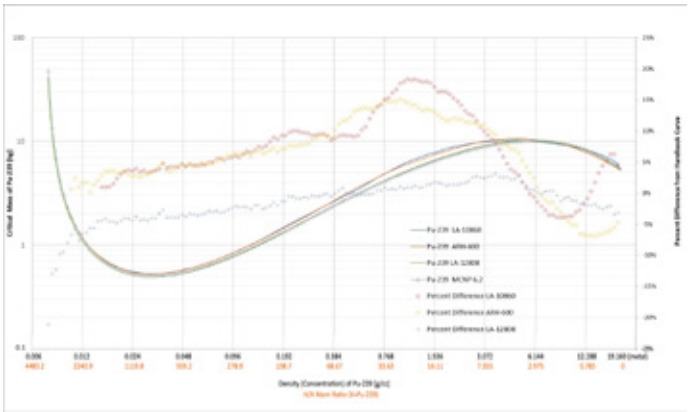


Figure 3. Sensitivity of keff to mass near criticality and average reactivity worth per unit mass for HEU and 239Pu

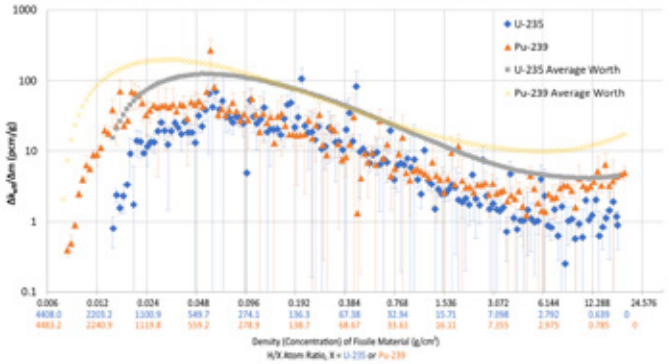


Figure 4. Critical and subcritical mass curves for ^{235}U as a function of ^{235}U concentration

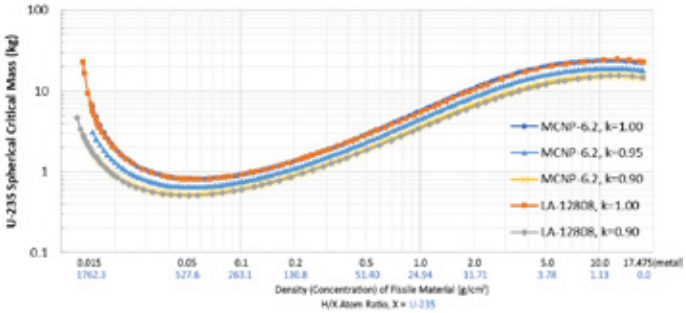


Figure 5. Sensitivity of keff per unit lethargy to the fission cross-section of ^{235}U as a function of energy and concentration of ^{235}U in a metal-water mixture

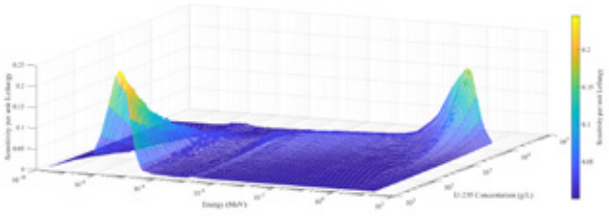
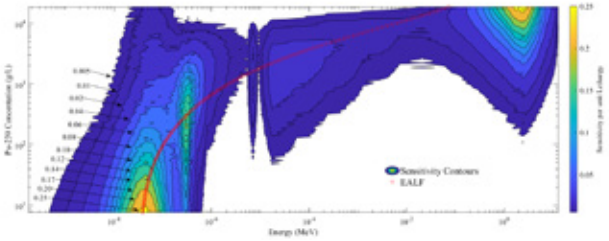


Figure 6. Contour plot of sensitivity of keff per unit lethargy to the fission cross-section of 239Pu with EALF plotted in the energy-concentration plane



TOOLS FOR VALIDATION AND UNCERTAINTY QUANTIFICATION WITH ANSWERS SOFTWARE

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The MONK® categorisation scheme has long been available to MONK users to assist in the choice of appropriate validation experiments for a chosen application. This has recently been supplemented by the production of a similarity index which indicates how similar an experiment is to the chosen application in terms of the sensitivity of the multiplication factor to the nuclear data. These tools have been used to select appropriate experiments for the OECD NEA wetted MOX powder benchmark. The results indicate that it is essential to use expert judgement in addition to the tools to ensure that appropriate experiments are chosen.

This paper also describes the production of a prototype Bayesian updating validation tool for use with ANSWERS® software. This is also applied to the OECD NEA benchmark on wetted MOX powders and the results for the first six cases are described. It is shown how the validation tool can be used to refine the estimate of the neutron multiplication factor and its uncertainty. For the six application cases described it is shown that up to a factor of two reduction in the estimated uncertainty can be obtained by the use of validation data.

EVALUATING SENSITIVITY-BASED SIMILARITY METRICS BETWEEN APPLICATIONS AND BENCHMARKS

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The Whisper-1.1 statistical analysis code targeted for nuclear criticality safety validation studies was released as part of the MCNP6.2 code release in 2018. Of the three primary functions that Whisper performs enroute to calculating an application baseline upper subcritical limit, this work primarily studies the details of the benchmark selection process. From a catalogue of ICSBEP experiments included with Whisper, it is necessary to select which are most similar to an application of interest. Presently, the most common sensitivity/ uncertainty-based metric to help analysts in assessing similarity between an application and a benchmark model is the correlation coefficient, c_k. This quantity provides a metric for how the nuclear data induced uncertainty between the models are shared and alike. Because the c_k between an application and benchmark is reliant and sensitive to the nuclear data covariance information, further investigation into the properties and behavior of c_k is the topic of this paper. A fine-grained view into the c_k metric, how it is computed, which quantities most contribute to it, and potential alternative metrics to be aware of are discussed in detail.

RESONANCE PARAMETERS AND COVARIANCE EVALUATIONS FOR THE GADOLINIUM ISOTOPES

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Gadolinium (Gd) isotopes are widely encountered in criticality safety and reactor applications. They are listed in the APPENDIX B of the US Department of Energy, Nuclear Criticality Safety Program (NCSP). The Institut de Radioprotection et de Sûreté Nucléaire (IRSN) works close to NCSP to address data issues pertinent to criticality safety. New evaluation of Gd isotopes, ^{152}Gd, ^{154}Gd, ^{155}Gd, ^{156}Gd, ^{157}Gd, ^{158}Gd, and ^{160}Gd, in the resolved resonance region has been performed with the code SAMMY. The resonance evaluations of these isotopes are in response to issues in connection to critical benchmark results and reactor physics analysis of power reactors. Furthermore, one of the main objectives is to understand and to generate more accurate cross sections and to improve their uncertainties and correlations thru the cross-section covariance representation. Both the resonance parameter and resonance parameter covariance are derived by a combination of the codes SAMMY, SAMDIST, and SAMINT. SAMMY is basically a differential evaluation tool based on the R-matrix theory for the cross section; SAMDIST verifies the statistical properties of resonance parameters while SAMINT permits the incorporation of integral data in the evaluation process.

The differential data used in the evaluation are transmission and capture cross section data available in the EXFOR database whereas critical benchmarks were those accessible in the ICSBEP database.

NEUTRON NUCLEAR DATA MEASUREMENTS AT GELINA

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[3] Pohang Accelerator Laboratory, Pohang University of Science and Technology, Pohang, Gyeongbuk 37673, Republic of Korea

[4] Department of Physics, Sungkyunkwan University, Suwon 16419, Republic of Korea

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Experimental work at the Geel Electron Linear Accelerator facility (GELINA) is dedicated to improve the accuracy of nuclear data, driven by observed deficiencies in evaluated data files. In this paper we will describe the experimental facilities and will give results of experiments of the total and capture cross section of silver, rhodium and gadolinium and we will compare the so derived data with calculations based on resonance parameters available in the literature.

IMPROVING NUCLEAR DATA LIBRARY PREDICTABILITY BY ACCOUNTING FOR TEMPERATURE EFFECTS USING RESONANCE PARAMETERS

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The ability to accurately model neutronics is key to the design of safe nuclear systems. The proper assessment of the impact that nuclear data covariance has on the uncertainty of integral parameters is essential for establishing safety margins. In the methods used for the propagation of uncertainty today, two major approximations are made for ease of computation: cross sections are energy-condensed (multigroup) and their covariance data is treated as temperature invariant. The accurate handling of the temperature dependence of covariance data represents a largely unstudied area. The impact of this effect on integral parameter uncertainty is unkown in magnitude and direction. To attempt to gauge this effect, the lowest lying s-wave resonance of ^{238}U is analyzed and its uncertainty as a function of temperature determined through random sampling of its parameters. One path forward for addressing this issue is by use of a windowed multipole parameter (WMP) library and sensitivity methods in Monte Carlo simulation. WMP is a resonance based representation of cross sections that allow for convenient on-the-fly Doppler broadening. A full WMP library has been developed for cross section reconstruction, but work on the development of a functional corresponding covariance library is underway. Some proposals in this effort are discussed.

DEVELOPMENT OF A GENERALIZED LATTICE SYMMETRY FORMULATION FOR THERMAL SCATTERING LAW ANALYSIS

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The thermal scattering law (TSL) for a given material is expressed as $S(\alpha,\beta)$, where α and β represent dimensionless momentum and energy transfer variables, respectively. The TSL is typically calculated using various approximations including the incoherent approximation and assuming cubic lattice symmetry for the material. The cubic approximation is useful for TSL analysis as it collapses the lattice information, utilizing an effective phonon frequency spectrum (i.e., density of states) as the fundamental input for calculating $S(\alpha,\beta)$, particularly under the incoherent approximation. In this work, the formulism of $S(\alpha,\beta)$ is revisited to eliminate the assumption of cubic symmetry and to allow the treatment of the actual symmetry of the lattice. In this case, capturing the directional dependencies for the lattice, given by the material's frequency dispersion curves and polarization vectors, accurately represents the material's dynamic properties. Furthermore, the developed treatment removes the atom site approximation as each atom is explicitly included into the calculation. To use in TSL evaluations, the generalized formulation has been implemented within the FLASSH: Full Law Analysis Scattering System Hub. It is applied to the self (i.e. incoherent) scattering law, $S_s(\alpha,\beta)$, which is typically tabulated in ENDF File 7 libraries. The application of this formulation is especially important for materials where marked deviations from cubic symmetry are expected and observed in the calculated Debye-Waller matrix (e.g., graphite, silicon dioxide, etc.). The resulting generalized cross sections show deviations from the cubic analysis up to 11.8% for graphite, a strongly non-cubic material, which has the potential to impact thermalization and therefore criticality safety calculations. If needed, the developed formulation is also applicable in the Doppler cross section treatment for materials with low energy absorption resonances.

PROGRESS ON ^{140,142}CE NEUTRON CROSS SECTION RESOLVED RESONANCE REGION EVALUATIONS[1]

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Oak Ridge National Laboratory is working on completing the resonance parameter evaluation of ^{140,142}Ce in the neutron energy range up to 200 keV. Performed with the support of the US Nuclear Criticality Safety Program, this evaluation aims to generate high-fidelity cerium cross-section and covariance data. A point-wise representation of the cross sections derived from the resonance parameters will provide improved calculations of self-shielding factors for nuclear criticality safety applications and additional evaluation support for continuous-energy radiation transport methodologies. The evaluation procedure uses the Reich Moore approximation of the R-matrix theory implemented in the SAMMY code system to fit resonance parameters to high-resolution transmission and neutron capture measurements of ^{nat}Ce performed in 2016 by the JRC-GEEL instrument scientists at the Geel Linear Accelerator facility as well as other experimental data sets on both natural and highly-enriched cerium samples available in the experimental library EXFOR. In the analyzed energy range this work aims to improve and extend the resolved resonance region present in the latest US nuclear data library ENDF/B-VIII.0 for ^{140,142}Ce isotopes. This paper will present the preliminary results of the R-matrix analysis based on recently measured ^{nat}Ce transmission data.

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Session 3 > Room 1

9h00 - 10h40 > Track 1

VALIDATION OF THE BURN-UP CODE MOTIVE USING ENDF/B-VIII DATA

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For determination of the nuclide inventories GRS has recently developed the 3D burn-up code MOTIVE (MOdular Tool for InVEntory Calculation). It iteratively couples a Monte-Carlo neutron transport code with continuous energy cross sections for flux calculation to a depletion code for nuclide inventory determination. In preparation of the deployment of the code to burn-up credit applications, validation calculations with MOTIVE are presented here using the code OpenMC for neutron flux determination and applying the latest ENDF/B-VIII nuclear data library. For this purpose, a set of over 70 PIE samples taken from the SFCOMPO2.0 data base of openly available PIE data hosted at OECD/NEA were analysed. The resulting inventory data are compared to experimental data in the form of C/E-1 values and are statistically analysed. Furthermore, an analysis of trends

against burn-up is performed. The results are also compared to a previous analysis in which similar validation calculations have been performed with MOTIVE using KENO-VI for neutron flux calculation in conjunction with ENDF/B-VII.1 data

INTERPRETATION OF GEDEON-1 AND GEDEON-2 GADOLINIUM DEPLETION EXPERIMENTAL ANALYSIS WITH THE DARWIN2.3 PACKAGE

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In the eighties a two phases experimental program, called GEDEON, was designed at CEA for the validation of gadolinium depletion calculation in a 13 x 13 LWR assembly (3.25% ^{235}U enrichment). In the first experiment, gadolinium pins are enriched with 3.5% of ^{235}U and contained 5% in mass of natural gadolinium. In the second experiment gadolinium support pins is made of depleted uranium (0.2% ^{235}U) and contains 8% in mass of natural gadolinium. DARWIN is an evolution code package developed at CEA in corporation with industrial companies (EDF, Framatome, Orano) in order to compute physical quantities of radioactivity related to various application fields such as nuclear fuel cycle. This paper presents the first interpretation, with DARWIN2.3 package, of those two experiments in the aim of gadolinium depletion experimental validation of DARWIN. Interpretations are performed using the European evaluation file JEFF-3.1.1 and a 281 group energy mesh. Results show quite good agreements between calculation and measurements, in respect with experimental and calculation uncertainties, especially for ^{155}Gd and ^{157}Gd isotopes depletion.

VERIFICATION AND VALIDATION OF THE DEPLETION CAPABILITY OF THE HIGH-FIDELITY NEUTRONICS CODE NECP-X

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The depletion capability of the high-fidelity neutronics code NECP-X is developed in this work. Two depletion libraries are developed for the depletion calculations. One full-fidelity depletion library contains 1547 isotopes. In order to reduce the computational burden and memory requirement, another new compressed depletion library is generated for NECP-X based on the full-fidelity deletion library. Neutron-induced cross sections from the EAF-2010 activation library are integrated into the compressed depletion library, which gives cross sections for multiple reaction channels not included in the ENDF library. The VERA depletion benchmark problems are utilized to assess the accuracy of the compressed depletion library, the new transport-depletion coupling method and depletion calculation of NECP-X by comparing eigenvalues and the pin power distributions. With the help of full-core high-fidelity depletion capability of NECP-X, the equivalent pin-cell model and assembly model, which are often used when predicting the isotope compositions for the spent nuclear fuel, are asserted by comparing to the full-core model. It shows that the corner rod and the fuel rod closed to the poisons should be prior to use the assembly model, while equivalent pin cell model is accuracy enough for the side fuel. Finally, measurement data for 22 fuel samples of Pressurized Water Reactors (PWR) are used to validate the depletion capability of NECP-X. The results demonstrate the good accuracy of NECP-X in predicting eigenvalues, the pin power distributions and the spent nuclear fuel isotope compositions.

HIGH FIDELITY KCODE MODELING OF SUBCRITICAL BENCHMARKS USING MCNP 6.2

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Several recently developed subcritical benchmarks system have eigenvalues that are is close to critical. These cases have been added to the International Criticality Safety Benchmark Project Repository. MCNP 6.2 simulations of these benchmark cases are available using MCNP's fixed source mode, but produce prohibitively long run-times for systems that are nearly critical. This work modifies MCNP 6.2 to enable the use of MCNP's higher fidelity fission physics while running in KCODE mode. KCODE causes a significant runtime reduction for each case, but it is of note that the K-Eigenvalue that is calculated by these systems can differ significantly from that of the benchmark. Also in ensuring that the number of events written to the PTRAC file is the same as for a fixed source case cane be difficult. This is important because PTRAC events are directly correlated to the singles count rate, a benchmarked quantity, of detectors. This work looked at several benchmarked cases from the ICSBEP report to determine the K-Eigenvalue accuracy and the runtime decrease for cases that were close to critical. The change in the K-Eigenvalue was close to fifty cents of reactivity using FREYA for GODIVA when comparing to the standard MCNP 6.2 benchmark. This was the largest difference with the FREYA and CGMF models being half of that. The cause of this may be due, in part, to the change in the fission multiplicity. This change however has shown a runtime decrease of 40% (44 minutes per case) increase for a 10$ subcritical system. This was larger, as expected, than the 12% (5.4 minutes per case) runtime decrease of the 21$ subcritical system.

FUNDAMENTAL PHYSICS SUBCRITICAL NEUTRON MULTIPLICITY BENCHMARK EXPERIMENTS USING WATER MODERATED HIGHLY ENRICHED URANIUM FUEL

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Five subcritical benchmark experiments with multiplication ranging from approximately 2 to 10 were carried out in order to provide validation of neutron transport software, nuclear data libraries, and neutron correlation techniques for multiplicity estimation. The experiments were conducted with the Inherently Safe Subcritical Assembly (ISSA), a highly enriched uranium (HEU) system moderated and reflected by water. The time-tagged list-mode neutron detection data for each of the five experiments was recorded and analysed to calculate the normalized doubles counting rates, R_{2F}. Each experiment was then modelled in COG and MORET to produce simulated list-mode neutron detection data. This data was then analysed by the same methods and compared to the experimental data to validate the neutron transport simulation software. Use of the Fission Reaction Event Yield Algorithm (FREYA) to generate correlated fission neutrons was found to significantly improve the accuracy of the simulations. The results of these experiments will be included in the 2019 International Criticality Safety Benchmark Evaluation Project (ICSBEP) handbook.

SUB-CRITICALITY MONITORING SYSTEM FOR THE RETRIEVAL OF FUEL DEBRIS IN FUKUSHIMA DAI-ICHI NUCLEAR POWER PLANTS

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During the retrieval of molten fuel debris in the Fukushima Daiichi Nuclear Power Plants (1F), the sub-criticality monitoring of the fuel debris is required in order to keep exposure doses to public and retrieval operators as low as reasonably achievable. Environmental conditions for sub-criticality monitoring systems in the 1F site are vastly different from those for monitoring systems in ordinary nuclear-fuel handling facilities in the viewpoints of environmental dose and the uncertainty of monitoring targets. There is little information for geometry, fuel composition and the ratio of hydrogen to heavy metal and so on. Therefore, robustness for the high dose condition and the uncertainty of the fuel debris has been required for developing the sub-criticality monitoring system. The neutron source multiplication method is well known as a quick method for the sub-critical monitoring. However, this method requires the reference condition of neutron-multiplication factor of monitoring target. As above mentioned, it is hard to acquire such detailed information for the 1F site. Thus, the neutron source multiplication method is fragile to the uncertainty. On the other hand, the reactor noise method is a robust method for monitoring sub-criticality to the uncertainty of the monitoring targets. However, this method requires a detail neutron-time-distribution and a long-time measurement to get statistically meaningful data. Therefore, we have been developing the prototype system by combining the neutron source multiplication method as the real-time monitoring method with the reactor noise method as the reference monitoring method to make the reference neutron-multiplication factor. This system is composed of plural detectors, current-type amplifiers, time-counter modules, a PC for system control and so on. A prototype system of sub-criticality monitoring for the 1F fuel-debris was tested by using ^{252}Cf and/or Am-Be neutron sources in the Kyoto University Critical Assembly (KUCA). In this paper, the summary of the system and the test results are presented.

VALIDATION OF MCNP®[1] ROSSI-ALPHA CALCULATIONS USING RECENT MEASUREMENTS

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This work will focus on the correct usage and validation of the MCNP KOPTS, kinetics options, card to obtain Rossi-α, or prompt neutron decay constant, for a system. The validation will focus on recent prompt neutron decay constant measurements completed at the National Criticality Experiments Research Center (NCERC) with comparison to both benchmark level and best working MCNP input decks. Measurements performed on Polyethylene Class Foils, HEU Zeus, HEU/Pb Zeus, IEU/Pb Zeus, KRUSTY, and Jupiter will be compared to their respective calculations. The prompt neutron decay constant measurements cover an energy spectrum of fast to thermal and two major fissionable isotopes, uranium and plutonium, with different reflectors and interstitial materials. For each of these experiments, prompt neutron decay constant measurements are performed on several subcritical

configurations. The Rossi-α, or prompt neutron decay constant at delayed critical, is always extrapolated from subcritical measurements, but in some cases direct measurements of the prompt neutron decay constant are taken at delayed critical. For the simulations performed in this work, the best available input deck is used to calculate the value of the prompt neutron decay constant at delayed critical. This paper will stress the importance of closely matching the critical configuration of the experiment when obtaining a result from the KOTPS card. This is due to the method in which the code calculates the prompt neutron decay constant. An explanation of this methodology is included. Based on the results included in this work, the KOPTS card in MCNP has a tendency to over-predict the prompt neutron decay constant by about 10%.

[1] MCNP® and Monte Carlo N-Particle® are registered trademarks owned by Triad National Security, LLC, manager and operator of Los Alamos National Laboratory. Any third party use of such registered marks should be properly attributed to Triad National Security, LLC, including the use of the designation as appropriate. For the purposes of visual clarity, the registered trademark symbol is assumed for all references to MCNP within the remainder of this paper.

CONVERSION FROM PROMPT NEUTRON DECAY CONSTANT TO SUBCRITICALITY USING POINT KINETICS PARAMETERS BASED ON A- AND K_{EFF}-EIGENFUNCTIONS

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The conversion from the measurement value of prompt neutron decay constant α to the subcriticality - ρ is investigated, using special point kinetics parameters which are defined by mixing forward α- and adjoint κ_{eff}-eigenfunctions. The mixing α-κ weighted point kinetics parameters are very useful to reduce the conversion bias of neutron multiplication factor in a deeper subcritical system. In the same manner as the k-ratio method, the mixing α-κ weighted point kinetics parameters can be well approximated by three eigenvalues of α, κ_{eff}, and prompt κ_{effp} to be simply estimated by the forward eigenvalue calculations without any adjoint calculations.

14h00 - 16h05 > Track 9

DETECTION OF A SLOW KINETIC CRITICALITY ACCIDENT BY THE RADIATION PROTECTION MONITORING SYSTEM

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Criticality accident detectors such as CAAS allow immediate evacuation of the staff in the event of a fast kinetic criticality accident with an inserted reactivity greater than $1. However, it is conceivable to have criticality accidents with low reactivity inserted, with neutron and gamma dose rates lower than the detection thresholds of criticality accident detectors. In these types of events a large dose rate is possible and it is necessary to guarantee the shelter of the staff. The Melox plant has a system for monitoring neutron and gamma dose rates in order to allow the sheltering of personnel in an airlock near the workstations in the event of a low-level radiological event. Thus, the staff is trained to evacuate the Room concerned in case of abnormal dose rate. It was determined the maximum accident leading to a lack of detection by the CAAS, taking into account the distances between the placet of the accident and the detection probes.This maximum accident triggering a radiation protection alarm, the workers would be present in the airlock and would be exposed to the radiation of the accident for the duration of an analysis conducted by the radiation protection service. It has been shown that in case of a maximum slow kinetic accident, the order of evacuation of staff would be given before an integrated dose of 100 mSv is reached.

CRITICALITY ACCIDENT ALARM SYSTEM ANALYSIS USING MCNP6.2 CONSTRUCTIVE SOLID GEOMETRY/UNSTRUCTURED MESH HYBRID

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Criticality Accident Alarm System (CAAS) design, analysis, and simulation require a combination of methodologies and techniques used for both criticality and shielding problems. MCNP6.2 is a general-purpose Monte Carlo radiation transport package with continuous-energy neutron and photon physics [1] making it suitable for CAAS analysis and simulation. Use of MCNP6 for analysis of criticality accident alarm systems has typically been done by building constructive solid geometry (CSG) models that may include extensive effort spent in specification. The MCNP6 unstructured mesh (UM) capability incorporates a way to specify geometries and conduct simulations on an unstructured mesh, which can be efficient when building complex geometries such as those relevant to CAAS analysis for large, complex facilities. This paper discusses a method that allows users to build a solid geometry or import existing computer aided drawing/computer aided engineering (CAD/CAE) models to create an unstructured mesh for use in MCNP6.2 calculations. Traditional CSG may be combined with UM for embedding an unstructured mesh representation of a geometry

and legacy CSG [2] to create a hybrid geometry. An advantage of the CSG/UM hybrid method is the ability to build the facility in the unstructured mesh and specify the critical cells and alarm system detectors in CSG.

In this work a constructive solid geometry/unstructured mesh hybrid model is created for use with MCNP6.2. The model is chosen for appropriateness to demonstrate the MCNP6 CSG/UM capability for solving problems relevant to CAAS analysis and for comparison to traditional CSG models used for analyzing criticality accident alarm systems. CAAS problems typically require the use of variance reduction techniques [3]. Methods designed to increase the efficiency of the calculations are discussed. Deterministic variance reduction methods available in Atilla10.0 are applied to the hybrid CSG/UM model [4]. Monte Carlo/deterministic hybrid variance reduction, in which FW-CADIS (forward-weighted consistently adjoint driven importance sampling) is used on the mesh in a deterministic calculation to generate weight windows and source biasing parameters which are applied in the MCNP6.2 simulations [5, 6, 7].

Results for CAAS analysis are presented and compared for MCNP6.2 unstructured mesh, constructive solid geometry, and unstructured mesh/constructive solid geometry hybrid models. Visualization of MCNP6.2 results, including use of the MCNP6 UM elemental edit output files and FMESH tallies over the geometry under consideration [8,9], is presented.

[1] C. J. Werner, et al., "MCNP User's Manual, Code Version 6.2," Tech. Rep. LA-UR-17-29981, Los Alamos National Laboratory, Los Alamos, NM USA (2017).

[2] R. L. Martz, "The MCNP6 Book on Unstructured Mesh Geometry: A User's Guide for MCNP6.2," Tech. Rep. LA-UR-17-22442, Los Alamos National Laboratory, Los Alamos, NM USA (2017).

[3] B. C. Kiedrowski, "MCNP6 for Criticaltiy Accident Alarm Systems – A Primer", Tech. Rep. LA-UR-12-25525, Los Alamos National Laboratory, Los Alamos, NM USA (2012).

[4] G. A. Failla, "Attila4MC – Application Example: Weight Windows Variance Reduction using CADIS and FW-CADIS. Varex Imaging, Gig Harbor, WA, USA (2019).

[5] D. Peplow, T. Evans, J. Wagner, "Simultaneous Optimization of Tallies in Difficult Shielding Problems", Nuclear Technology, 163, pp. 3 (2009).

[6] J. Wagner, E. Blakeman, and D. Peplow, "Forward-Weighted CADIS Method for Variance Reduction of Monte Carlo Calculations of Distributions and Multiple Localized Quantities". International Conference on Mathematics, Computational Methods & Reactor Physics, Saratoga Springs, NY (2009).

[7] T. Miller and D. Peplow, "Guide to Performing Computational Analysis of Criticality Accident Alarm Systems", Oak Ridge National Laboratory, Oak Ridge, TN, USA (2103).

[8] J. A. Kulesza, "A Python Script to Convert MCNP Unstructured Mesh Elemental Edit Output Files to XML-based VTK Files," Tech. Rep. LA-UR-19-20291. Los Alamos National Laboratory, Los Alamos, NM USA (2019).

[9] C. J. Solomon, C. R. Bates, and J. A. Kulesza, "The MCNPTools Package: Installation and Use," Tech. Rep. LA-UR-17-21779. Los Alamos National Laboratory, Los Alamos, NM USA (2017).

MAVRIC-SCALE SEQUENCE FOR CRITICALITY ALARM SYSTEM APPLICATIONS

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The Juzbado Fuel Fabrication Plant has a Criticality Alarm System (CAS) covering every area where nuclear material is handled. Every CAS detector shall cover a circular surface with a maximum radius of 36,5 or 150 m depending on the criticality risk. This surface will be smaller if attenuations between the nuclear material and the CAS detectors can be considered. In case of changes on the fabrication lay-out, it is necessary to re-calculate the transmission factors to analyze their impact on the CAS detectors distribution.

The SCALE package contains the MAVRIC control module that provides a tool for shielding and radiological protection calculations. Before using the MAVRIC module to obtain the transmission factors, it has been validated against two different methods:

Method 1, which follows the Regulatory Guide 3.34.

Method 2 based on the dose half-value layers for concrete as a function of the energy spectra obtained from reference data.

These two methods were compared to a MAVRIC simulation which consists of modeling a 20 cm spherical shape source with the energy spectra used for the CAS design (400 to 7100 keV). The results obtained by these three methods are statistically equivalent. Therefore, it is concluded that the MAVRIC sequence is a useful tool to calculate transmission factors for a source spectra in the range of the CAS design.

In this paper, we present the study carried out to obtain the transmission factors of the concrete walls and slabs used in the Plant's building, from which the effective radii of coverage are detached in case of attenuations between the material and the CAS detectors. This study allows to analyze the impact of Plant's design changes on the distribution of the CAS detectors, and to optimize the locations of the CAS detectors to meet the regulatory requirements.

Finally, with the aim of verifying the margin of safety of the assumptions made in the distribution of the CAS detectors and their coverage areas, a criticality event is simulated. The results of this simulation conclude that in the event of a criticality accident, it would be detected by the CAS detector installed in the area where the accident occurs and by the detectors placed in the adjacent areas.

THE CAAS-3S NEXT-GENERATION CRITICALITY ACCIDENT ALARM SYSTEM

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The CAAS-3S is a next-generation criticality accident alarm system designed for facility operations over the next several decades. The system monitors areas where a criticality excursion can potentially take place and alarms rapidly for the prompt

evacuation of personnel. We present an overview of the system design considerations, testing to criticality standards, installation and operational considerations, and benefits to plant operators. Although the CAAS-3S is newly designed with a view to the future, it incorporates over 40 years of operational excellence obtained from previous models within the product. The detection probe design is based on the highly reliable analog signal chain used in the previous EDAC-xx models which have had successful safety records and very low false alarm rates. A key achievement of the design is the seamless integration of the safety function (based on the proven probe design) with the many supporting functions demanded by modern facilities. The use of modern programmable logic controls (PLCs) and client-server supervision software enables convenience, flexibility, and long-term sustainability while meeting the critical function of the system.

PRESENTATION ON THE FUTURE CRITICALITY INCIDENT DETECTION SYSTEM AT AWE

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A prototyped IS820 Criticality Incident Detection System (CIDS) was successfully designed and produced by an internal design team at AWE. A conformity project was also undertaken, where a number of tests and assessments were carried out to ensure compliance with the specified AWE requirements along with international/domestic standards.

The license to manufacture the future IS820 CIDS was secured by Ultra Electronics® which allows the system to be manufactured on a larger scale and marketed as a Commercial of the Shelf (COTS) product which is available to AWE. Ultra Electronics® can also sell the system on the world market on behalf of the MOD & AWE.

Session 4 > Conference Room LOUIS ARMAND

9h00 - 10h40 > Track 7

A MISLOAD ANALYSIS METHODOLOGY SUPPORTING CRITICALITY ANALYSIS OF SPENT NUCLEAR FUEL CANISTERS USING AS-LOADED CONFIGURATIONS[1]

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This paper presents an assembly misload analysis methodology developed to support criticality safety analysis of spent nuclear fuel (SNF) dual-purpose canisters (DPCs) using as-loaded configurations. The misload analysis approach is based on Interim Staff Guidance 8 "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks" (ISG 8 rev. 3) [1], extended to support as-loaded configurations. The two misload scenarios analyzed included (1) the correct assemblies are selected from the pool but placed incorrectly into the most reactive configuration inside the canister, and (2) the incorrect, most reactive assembly/assemblies in the pool is/are placed into the most reactive position(s) in the canister. Results from this misload analysis approach can be combined with misload probability to support criticality safety assessment of SNF during storage, transportation, and disposal. The misload analysis has been applied to 67 loaded DPCs at two sites using a disposal scenario. The analyzed sites include pressurized and boiling water reactors and two canister variants—MPC-32 and MPC-68.

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[1] Division of Spent Fuel Storage and Transportation, Interim Staff Guidance – 8, Revision 3, Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks, US Nuclear Regulatory Commission (2012).

CRITICALITY SAFETY ANALYSIS OF SPENT NUCLEAR FUEL CANISTERS USING AS-LOADED CONFIGURATIONS[1]

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Dual purpose canisters (DPCs) used for storage and transportation of spent nuclear fuel (SNF) are typically designed and evaluated using bounding (enveloping) fuel characteristics such as fuel type, fuel dimensions, initial enrichment, discharge burnup, and cooling time. This is a design basis, bounding licensing approach for SNF storage and transportation systems, as licensing and

supporting safety analysis reviews are performed prior to the actual fuel loading. The bounding fuel characteristics for a system are developed by fully utilizing the safety limits required or recommended by the regulators such as neutron multiplication factor (k_{eff}) approaching 0.95 to maximize the system utilizations. In reality, there are wide variations in SNF assembly burnups, initial enrichments, and cooling times. Therefore, dry storage systems are typically loaded with assemblies that satisfy the bounding fuel characteristics defined in the certificate of compliance (CoC), with some amount of unquantified and uncredited margin. This paper presents an as-loaded canister-specific criticality analysis approach for quantifying inherent uncredited margins in already loaded DPCs. The as-loaded analysis approach has been implemented in a new SNF management and analysis tool - The Used Nuclear Fuel-Storage, Transportation, and Disposal Analysis Resource and Data System (UNF-ST&DARDS). The paper presents as-loaded criticality analysis of 616 canisters at 28 US reactor sites. Additionally, the paper discusses an as-loaded analysis methodology validation approach using detailed reactor operational and fuel design data.

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ON THE BENEFIT OF FAST-NEUTRON REACTOR FUEL DEPLETION FOR TRANSPORTATION

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In recent years, some efforts have been devoted in France to research projects for handling Phenix Fast-neutron Reactor (FR): wet storage, reprocessing process and transportation. This paper assesses the benefit of taking account of the depletion of FR fuel elements for transportation.

After a brief presentation of the calculation tools and models for depletion and criticality calculations, the depletion and criticality options are discussed. Then, interesting results show that the use of a low burnup level may allow an optimised loading of FR fissile fuel assemblies in the TN® 17/2 transport cask. Furthermore, the results for FR fissile assemblies also encompass criticality calculations for FR fertile assemblies. Indeed, due to TN®17/2 transport cask design, substantial margins are available for FR fertile assemblies, even for strongly conservative hypotheses on fuel inventory. Consequently, considering their irradiation is not of interest for this case, but could be for other transportation configurations. Finally, the paper focuses on depletion codes validation, a key element for a straightforward and effective implementation of this approach.

CRITICALITY SAFETY ANALYSIS FOR STORAGE AND TRANSPORTATION APPLICATIONS USING NRC ISG-8 REV. 3

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Burnup credit methodology takes credit for the reduced reactivity due to the irradiation of a fuel assembly when performing criticality safety analysis. The reduction of reactivity with fuel burnup is mainly due to the net reduction of fissile nuclides and the production of neutron absorbing actinides and fission products. The output of a burnup credit analysis is a criticality loading curve that defines the minimum required burnup as a function of initial enrichment.

Because burnup credit requires burned fuel isotopes as input, the uncertainty in computed isotopes and the effect on reactivity must be quantified. The isotopic bias and bias uncertainty values are expressed in units of Dk and included as a penalty when computing k_{eff}. This paper presents an example computation of isotopic bias and bias uncertainty based on the Monte Carlo uncertainty sampling method. U.S. Nuclear Regulatory Commission (NRC) Interim Staff Guidance 8 Rev. 3 (ISG-8) is the main framework for implementing PWR burnup credit for criticality safety analysis. ISG-8 includes default values for isotopic bias and bias uncertainty that may be used directly if certain conditions are met. The computed isotopic bias and bias uncertainty are compared against the ISG-8 values.

In addition to the criticality analysis, misload analyses may be performed in lieu of burnup measurements. A misload analysis addresses potential events involving the misplacement of fuel assemblies into a storage or transportation system that do not meet the proposed loading curve. A misload analysis considers the occurrence of a single severely underburned assembly and multiple moderately underburned assemblies.

DETERMINATION OF BOUNDING AXIAL BURNUP DISTRIBUTIONS FOR PWR SPENT FUEL ASSEMBLIES DISCHARGED FROM NUCLEAR POWER PLANTS IN SOUTH KOREA

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In this work, the bounding axial burnup profiles are evaluated for 12 burnup groups through the core following calculations and statistical analysis of k_{eff} values (or end effects) with criticality calculations for a spent fuel storage pool and 4,582 axial burnup profiles for the spent fuel assemblies discharged form KORI Unit 1, 2, 3 and HANBIT Unit 3. The results of the analysis showed that the average end effects for the burnup groups lower than 34~38 MWD/kg were estimated to be negative while the maximum end effects were estimated to be negative for the burnup groups lower than 22~26 MWD/kg and that the maximum positive and effects were ranged from 0.29% ~ 4.43% Dk depending on the burnup groups. For the high burnup groups, the bounding axial burnup profiles are resulted from the KORI Unit 3 spent fuels having natural uranium axial blankets and they have very low burnups in top and bottom end nodes while the bounding axial burnup profiles for intermediate burnup groups are resulted from the KORI Unit 2 spent fuels and they showed small end effects.

OVERVIEW OF THE RECENT BWR BURNUP CREDIT PROJECT AT OAK RIDGE NATIONAL LABORATORY[1]

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Oak Ridge National Laboratory (ORNL) and the US Nuclear Regulatory Commission (NRC) have completed a five-year program to investigate burnup credit (BUC) for boiling-water reactor (BWR) spent nuclear fuel (SNF) stored in storage and transportation systems. The project examined both peak reactivity BUC, also sometimes called gadolinium credit, and extended BWR BUC. Here extended refers to credit for burnups beyond that associated with peak reactivity. The findings related to peak reactivity BUC are summarized in NUREG/CR-7194, and four additional NUREG/CR documents present the results of the studies of extended BWR BUC. The first of these documents, NUREG/CR-7224, presents results of investigations to determine the effects of axial moderator density profiles, control blade use, and axial burnup profiles. Studies on the impact of core operating conditions and assembly-specific depletion conditions are addressed in NUREG/CR-7240, validation of depleted SNF isotopic predictions is addressed in NUREG/CR-7251, and validation of k_{eff} calculations for extended BWR BUC are discussed in NUREG/CR-7252. A summary of the entire project, including major conclusions regarding each of the studies, is included in this paper.

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BURNUP CREDIT IMPLEMENTATION FOR ENRICHED REPROCESSED URANIUM USED FUEL TRANSPORTATION

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Orano TN has implemented Burnup Credit (BUC) approaches for the demonstration of the sub-criticality for transport and dual purpose casks loaded with PWR uranium oxide (UO_2) used fuel assemblies. In the early 1980s, a simplified BUC method, based only on the major actinides and a low level of burnup credit, was applied. In 2000s, Orano TN developed and then implemented an advanced BUC methodology. The implementation of this method has the advantage of taking benefit from the negative reactivity due to not only major actinides but also a limited number of Fission Products (FPs). This new BUC approach is implemented since 2013 in different transport and dual purpose casks (TN® 24 E TN® 17/2, TN®17 MAX).

This advanced BUC approach requires, among other aspects, the definition of a bounding isotopic compositions of irradiated fuel to determine the maximum cask reactivity according to regulatory transport conditions. Therefore, depletion calculations of the fuel assemblies have to be performed with a validated computer code system, penalizing irradiation parameters and by taking into account a bounding initial composition of the fuel.

Usually, the BUC methodology uses Enriched Natural Uranium to determine the isotopic composition of the fuel after irradiation for criticality calculations. Nevertheless, Enriched Reprocessed Uranium (ERU) may be used for the manufacturing of the PWR UO_2 fuel assemblies. As far as criticality is concerned, the main difference between Enriched Natural Uranium and ERU, for the same U-235 content, is the presence of U-232, U-234 and U-236 in the ERU initial composition.

This paper presents sensitivity calculations to assess the impact on transport cask reactivity when ERU initial composition is used for BUC applications.

USING THE ORNL SPENT FUEL DATABASE TOOL UNF-ST&DARDS FOR AS LOADED AND SCOPING CALCULATIONS FOR THE SWEDISH SPENT NUCLEAR FUEL REPOSITORY

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During 2018-2019 SKB has investigated different types of fuel databases. The goal is to have a database with all essential fuel information necessary to perform criticality, decay heat, thermal and radiation shielding analysis as well as isotopic inventory. One important requirement for the database is that it should be possible to couple with analysis tools (e.g. Keno, Origen and MCNP) in a quality ensured way. Another aim is to be able to do "as loaded" calculations for our different storage and transportation systems. This will open the way to a more optimised and cost effective spent fuel operation. The realisation of this would be greatly simplified by an automatized way to import core follow data from the Swedish power plants. This paper will present initial results from evaluation of one of our options, the spent fuel database and analysis tool UNF-ST&DARDS developed by Oak Ridge National Laboratory (ORNL).

UNF-ST&DARDS is developed by ORNL to perform high quality data storage and analysis for the US used fuel management. SKB has evaluated the tool for the Swedish system and evaluated it against some SKB use cases such as nuclide variations for different operating histories, decay heat calculations and as-loaded criticality analysis for the disposal canister. The paper will present experiences from implementation of UNF-ST&DARDS for the Swedish case and a comparison between the regular criticality analysis and the as-loaded methodology used in UNF-ST&DARDS for the current disposal canister design. The preliminary results indicate that for most of the cases the margins are large, as expected.

UNF-ST&DARDS is a convenient tool for an operational organisation such as SKB where few persons have to be able to perform multiple types of analysis as it offers an automated way of running calculations with high quality through a graphical interface for non-experts. The easily accessible data and analysis capability encourage staff to investigate more which can lead to optimization of the whole system.

14h00 - 16h05 > Track 8

OPTIONS FOR DEMONSTRATING CRITICALITY SAFETY FOR GEOLOGICAL DISPOSAL OF UK SPENT FUEL

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A demonstration of the safe geological disposal of spent fuel must include consideration of criticality safety during transport of the fuel to the disposal facility, during disposal operations and after facility closure. This paper describes RWM's preferred solutions for demonstrating criticality safety for each of these phases of spent fuel management. Criticality safety can be ensured by implementing criticality controls, such as limits on fuel mass, controls on the arrangement of the fuel, inclusion of multiple water barriers, and inclusion of neutron poisons in container components. RWM's preferred solution for ensuring criticality safety during spent fuel transport operations is based on demonstrating that the transport package provides multiple barriers to water entry into the disposal container. The preferred solution for demonstrating criticality safety during the disposal facility's operational phase is based on application of a 'double contingency' approach. That is, a demonstration that a criticality accident cannot occur unless at least two unlikely, independent and concurrent changes in conditions specified as essential to criticality safety have occurred. Demonstration of the criticality safety of most types of spent fuel after closure of the disposal facility relies on credit being taken for fuel irradiation.

DERIVATION OF WASTE PACKAGE CRITICALITY CONTROLS THAT ENSURE THE LONG-TERM CRITICALITY SAFETY OF A UK GEOLOGICAL DISPOSAL FACILITY

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Radioactive Waste Management (RWM) is responsible for implementing geological disposal of the UK's higher-activity radioactive wastes. These wastes include large quantities of fissile nuclides and, therefore, demonstration of the criticality safety of the wastes under disposal conditions forms an important component of RWM's disposal system safety case.

RWM has developed a methodology for establishing criticality controls on waste packages that contain intermediate level waste (ILW) and application of this methodology will support the criticality safety demonstration for the geological disposal facility (GDF). The methodology is based on the UK regulatory requirement to demonstrate that post-closure criticality is not a significant concern, which means showing that such a criticality event is unlikely to occur and, if it did occur, it would be of low consequence to the performance of the GDF.

The GDF siting process in the UK is ongoing and the criticality safety assessment approach is currently based on consideration of illustrative GDF concepts. Necessarily, a cautious approach is taken to assessing post-closure criticality scenarios in order not to preclude any potential GDF concepts. This paper discusses how, even at the generic stage of GDF development, by working with waste packagers to understand ILW packaging concepts and characteristics, it is possible to develop waste package criticality controls that ensure that GDF post-closure criticality safety requirements can be met taking a cautious approach without imposing unnecessary radiological risks and costs in the waste packaging process.

Interaction with waste packagers takes place through RWM's waste package Disposability Assessment process that includes discussion, development and review of the criticality safety strategy for specific wasteforms through different stages of packaging concept development. The paper discusses how, through this process, criticality safety controls can be identified for potentially challenging wastes by focusing on how credit can be taken for specific properties of the wasteform and container in building criticality safety arguments. Recent examples of the application of the methodology are presented, where the long-term behaviour of particular components of the waste package (e.g., container, waste, encapsulant or immobilisation matrix) is taken into account when deriving post-closure criticality controls. The paper emphasises the importance of waste package records in this process as an evidence base for the criticality safety case that will need to be developed when a GDF site is available.

A GENERIC CRITICALITY SAFETY ASSESSMENT FOR THE GEOLOGICAL DISPOSAL OF WASTES PACKAGED IN SHIELDED CONTAINERS

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Radioactive Waste Management Limited (RWM) is the subsidiary of the Nuclear Decommissioning Authority (NDA) that is responsible for delivering a Geological Disposal Facility (GDF) and providing solutions for the management of higher activity radioactive waste in the UK. At present, a site for a UK GDF has not been identified; in order to progress the programme for geological disposal while potential disposal sites are being sought, RWM has developed illustrative disposal concept designs for different types of host rock.

The wastes requiring disposal include fissile nuclides and, therefore, demonstration of criticality safety under disposal conditions will form an important component of RWM's disposal system safety case for the GDF. In order to support the waste packaging process in ensuring that criticality safety of the GDF can be achieved, RWM is undertaking work to develop generic Criticality Safety Assessments (gCSAs) for a range of package designs and waste loading assumptions based on the illustrative disposal concept designs. This paper presents RWM's development of a gCSA for intermediate level waste (ILW) packaged in shielded containers.

Three types of container have been considered: 6 m^3 box, 2 m box and 4 m box. It is expected that such waste packages will mostly be transported as IP-2 transport packages and will meet one of the International Atomic Energy Agency's (IAEA's) criteria for the use of such waste packages, such as a fissile exception criterion. That is, the waste packages would be expected to contain relatively small masses or concentrations of fissile material, although the fissile material could include highly enriched uranium and plutonium as well as uranium of lower enrichments.

Even though it may be possible to transport the waste packages as IP-2 transport packages, the evolution of the waste packages during and after geological disposal requires assessment. Such assessments are necessary in order to identify any criticality controls that need to be adopted to ensure that GDF criticality safety requirements are met. Thus, the gCSA for shielded waste packages involved assessment of conditions during GDF operations and in the long term after GDF closure. For the GDF post-closure phase, criticality scenarios involving ingress of water into degrading waste packages in a disposal vault and migration and accumulation of fissile and other materials from stacks of waste packages in a vault have been assessed. Waste package fissile material limits have been derived based on these post-closure criticality scenario assessments as a function of the expected distribution of fissile material in the waste packages and the presence of any neutron moderating and/or reflecting materials. Calculations of the neutron multiplication factor for different scenario configurations have been undertaken using MCNP. This paper presents the key results of the gCSA.

ANDRA'S POST CLOSURE NUCLEAR CRITICALITY SAFETY ASSESSMENT TOWARDS THE LICENSING APPLICATION FOR CIGEO

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The French National Radioactive Waste Management Agency (Andra) has entered an industrial design development phase and is now preparing the overall safety case towards the licensing application for the "Centre industriel de stockage en milieu géologique (Cigéo)" (Industrial Center for Geological Disposal).

Cigéo represents more than 25 years of acquisition of scientific and technical knowledge and the development of safety methods appropriate to deep geological repository (DGR).

Andra has established early 2016, the Safety Options Files (namely "Cigéo 2015") to precede the license application. In that framework, operational and post-closure nuclear criticality safety assessments have been conducted and as such in a coordinated approach.

In accordance with the safety guide for the final disposal of radioactive waste in a deep geological formation, published by the French Nuclear Safety Authority (NSA) in 2008, "after the closure of the disposal facility, the protection of human health and the environment must not depend on institutional monitoring or control as there is no certainty that this can be maintained for more than a limited period".

Cigéo is therefore designed to robustly fulfil the post-closure safety functions in a passive way and specifically to prevent the criticality risk.

The geological disposal system consists of a set of manufactured and natural components including the Callovo-Oxfordian host rock formation.

The evolution on the long term will affect the geometry of the vault and waste packages and eventually the containment of fissile material. In that respect, the evolution of waste packages is considered as a whole in term of geometry and chemical evolutions of all components.

The aim in studying these phenomena is to verify that the successive evolutions of waste packages, vaults and fissile material according to phenomenological point of view are not likely to lead to a criticality of the system (k-effective strictly less than 1).

Each evolution of these parameters, which may have a favorable or unfavorable influence on reactivity, is studied: first, independently then in a combined manner in accordance with temporality.

The main parameters to be considered for the waste packages evolution during the post-closure phase for the nuclear criticality safety assessment are the chemical evolution of concrete, the chemical evolution of interstitial water in the disposal, the material thickness and the corrosion of metals.

Considering those phenomena, the assessment of the post-closure nuclear criticality risk is therefore characterized by taking into account mainly the chemical and mechanical evolutions of the vaults and waste packages.

In order to treat the various uncertainties, criticality calculations shall include conservatisms and the most penalizing evolutions.

The objective of the paper is to present methods, results and preliminary lessons learned at that stage about the assessments of the impacts of mechanical and chemical evolutions of the disposal system as such the waste packages and the fissile material in Cigéo.

THE CREDIBILITY OF POST-CLOSURE CRITICALITY: CONSIDERATIONS FOR MOX SPENT FUEL AND WASTES CONTAINING URANIUM-233 AT DISPOSAL OR FROM INGROWTH

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Radioactive Waste Management has been tasked to plan, build and operate the UK's Geological Disposal Facility (GDF). At present a site for a UK GDF has not been identified, and RWM has produced a generic Disposal System Safety Case (gDSSC) to put forward the safety arguments using a range of illustrative disposal concepts and host geologies.

Given that a GDF will include the disposal of fissile material which could, under certain conditions, lead to criticality, the demonstration of criticality safety forms an important part of the gDSSC. In particular, the UK environment agencies' Guidance on Requirements for Authorisation for a GDF requires a demonstration that "the possibility of a local accumulation of fissile material such as to produce a neutron chain reaction is not a significant concern", alongside a requirement to consider, as a 'what-if' scenario, the impacts of a postulated criticality event on GDF performance.

RWM has undertaken substantial research into the criticality safety of radioactive waste disposal, seeking to demonstrate that the likelihood and consequences of criticality after GDF closure are low for a wide range of fissile materials that may require geological disposal. This paper builds on the knowledge from previous criticality safety research to extend the understanding to fissile systems that could include uranium-233 (derived either from that present in the waste at the time of disposal or from later ingrowth), or MOX (mixed oxide) fuel in a fresh or irradiated state. The analyses for these additional systems show that the conclusions from previous work are applicable – i.e. for the underlying assumptions of the research, post-closure criticality is unlikely to occur, rapid transient criticality (characterised by a short-lived, but potentially substantial, energy release) is not credible and 'what-if' criticality events would not have a significant impact on the post-closure performance of a GDF.

Session 1 > Room A-B

9h00 - 10h40 > Track 5

REGULATING CRITICALITY SAFETY: THE EFFECT OF TEMPERATURE ON REACTIVITY

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The effect of temperature on reactivity is of intense interest to the criticality community. This paper describes, in the context of nuclear regulation, research commissioned by the UK nuclear regulator into the effects of temperature on reactivity.

Standard reactor physics theory shows that temperature variations could affect criticality safety margins. The neutron multiplication factor, k, of a system may increase or decrease with temperature. This is dependent on competing parameters; therefore, inferences drawn from the inspection of one system may not apply to another. There is currently little information regarding the reactivity change that may be expected for specific fissile systems below Room temperature.

The UK's nuclear safety regulator, the Office for Nuclear Regulation (ONR), expects dutyholders to assess the effects of temperature variation on criticality safety where appropriate. There are two areas of criticality safety where assessing temperature-related effects is important; namely, in transport applications for fissile materials packages, and in plant safety cases.

Until recently, it has not been practicable to assess the effects of temperature on criticality safety across a wide range of temperatures because the major codes and nuclear data libraries did not have information to produce calculations at some temperatures (e.g., below Room temperature). In the past, modelling conservatisms have been made to account for potential positive increases in reactivity with temperature. The latest criticality codes and data libraries allow criticality calculations at specific temperatures to be calculated.

To inform regulatory decision making, ONR commissioned research on the effects of temperature on system reactivity across a range of temperatures (including low temperatures) due to changes in cross section in isolation (i.e., no account of changes in material density). A technical support contractor was employed to explore the effect of temperature on reactivity in a range of infinite fissile systems (with varying fissile content, and degrees of moderation and absorption). The outcome was a report presenting calculations of reactivity trends over a range of temperatures, and discussing the important nuclear phenomena involved in the relationship between system temperature and reactivity.

This paper will describe the main findings of this report.

REGULATING CRITICALITY SAFETY: USE OF BURN-UP CREDIT IN THE ASSESSMENT OF CRITICALITY RISK

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This paper provides an overview and key findings of research into the use of burn-up credit in criticality safety assessment, and the potential implications of these findings for the regulation and management of criticality safety in Great Britain (GB).

In GB, the drop in reactivity that occurs when nuclear fuel is irradiated in a nuclear power reactor has typically not been claimed. Instead, criticality safety assessments have normally assumed that the fuel is unirradiated with no reduction in the fissile material present.

Although this 'fresh fuel' approach is conservative, it leads to an overestimation of the calculated neutron multiplication factor (k_{eff}) and may lead to additional operational burdens being placed upon dutyholders. In future, it is possible that GB dutyholders may take credit for the reduction in reactivity, known as 'burnup credit', that occurs when fissile material is consumed in a nuclear reactor.

However, accurately quantifying the effect that fuel burnup will have on reactivity is difficult as it is dependent upon many factors such as the reactor type, fuel isotopic composition assumed, irradiation history, density of the primary coolant, fuel assembly position in the reactor and the cooling time. Due to this complexity, the inventory and reactivity prediction calculations require simplifying assumptions.

In order to provide authoritative and independent information on the key aspects of burnup credit, the Office for Nuclear Regulation (ONR) commissioned a research project. This provided the latest knowledge and advice regarding aspects such as the various methods for modelling fuel burnup in the inventory / reactivity prediction calculations, the code / nuclear data validation available and the current technology for calculating / confirming fuel burnup and its reliability. The findings from this research project are to be used by both ONR in their regulatory duties when forming judgements regarding duty holder safety cases, and by duty holders themselves to better understand ONRs expectations.

This paper provides an overview of the key findings of this research and potential implications for the regulation and management of criticality safety in GB. A 'Regulator Question' set for use when assessing criticality safety cases is given that is intended to aid both regulator and duty holder in their assessments, which enables regulatory attention to be targeted proportionately on those areas of most importance.

IMPLEMENTATION OF FISSION PRODUCTS CREDIT FOR PWR MOX

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Preliminary studies on Burnup Credit for PWR MOX fuels have pointed out the difficulties related to the determination of a conservative isotopic composition to cover a wide range of MOX fuels. Indeed, contrary to PWR UOX fuels, for PWR MOX fuels, a fresh-fuel conservative isotopic composition does not inevitably lead to the most reactive situation after irradiation due to the amount of plutonium fertile isotopes (^{238}Pu, ^{240}Pu...).

Latest studies related to the use of Burnup Credit for PWR MOX fuels have shown that the fission products contribute to the reactivity decrease more significantly than the actinides for a cooling time of up to 5 years.

This observation allows to consider an implementation of Burnup Credit for PWR MOX fuels based on the reactivity worth of only 15 fission products (already identified for PWR UOX credit burnup applications): ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{103}Rh, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd, ^{147}Sm, ^{149}Sm, ^{150}Sm, ^{151}Sm, ^{152}Sm, ^{153}Eu and ^{155}Gd.

Several calculations were performed to assess the impact of several irradiation parameters on the reactivity worth of fission products (e.g. environment of MOX fuel assembly, control rods insertion, moderator density, fuel temperature, specific power, cooling time).

Moreover, further calculations were performed to verify that taking into account a fresh-fuel conservative isotopic composition leads to the most reactive situation after irradiation when Burnup Credit is based only on the reactivity worth of fission products.

The objective of the paper is to present the studies allowing to validate a methodology of Burnup Credit for MOX PWR fuels and to estimate the reactivity gain for a configuration of industrial interest.

THE OXNIT DENSITY LAW IN CRISTAL PACKAGE: AN EASY WAY TO PREDICT THE COMPOSITION OF DISSOLVED OXIDE IN NITRATE SOLUTIONS

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In the criticality-safety practice, density laws are a key point for determining the composition of fissile solution media. Various fissile materials are encountered in the fuel cycle from manufacturing of fuel pellets to reprocessing of fresh or used fuel. During fuel dissolution, uranium/plutonium oxide is assumed to coexist with uranyl/plutonium nitrate under various physical forms: uranium/plutonium oxide powder, uranium/plutonium pellets' dissolution residues, undissolved uranium/plutonium oxide particles in acidic nitrate solutions. To assess these configurations a dedicated density law (called "OXNIT") is therefore necessary to characterize the composition of the oxide-nitrate mixture in criticality studies. A first density law was programmed in a criticality-safety GUI. However, due to many constraints limiting its use by practitioners, this former OXNIT density law did not answer the needs for criticality-safety assessment.

Consequently, in 2017, IRSN and ORANO decided to propose a new law enlarging its use. This was done in the LATEC workbench [1], which is part of the CRISTAL V2.0 criticality safety package [2]. This law is based on the volume addition of a crystal of oxide in a solution of nitrate, whose density is determined using the "isopiestic" nitrate density law. Thanks to the LATEC framework, this law now meets the needs of users regarding the various parameters that can be taken into account in the criticality safety assessment. Thus, the OXNIT mixture is defined by the following parameters:

- Oxide physical form (the form of a crystal or a powder),
- Isotopic vector of uranium/plutonium oxide and uranyl/plutonium nitrate,
- U/Pu content in oxide and in nitrate; thus it is possible to simulate a faster dissolution of U in nitrate,
- Moderation expressed in H/X or in C(X),
- Accounting for acidity and poison, such as Gd.

By comparison with the previous version of CRISTAL V1 [3], a large flexibility is now offered to the criticality safety practitioners in the definition of the fissile media.

The object of the paper is to present the breakthroughs that were detailed previously and to show the validation that was done to guarantee the correctness of the obtained compositions.

[1] B.T. Rearden and M.A. Jessee, Editors, "SCALE Code System", ORNL/TM-2005/39, Version 6.2, Oak Ridge National Laboratory, Oak Ridge, TN (2017).

[2] OECD-NEA,"International handbook of evaluated criticality safety benchmark experiments," Tech. Rep. NEA/NSC/DOC(95)03, NEA Nuclear Science Committee, 2016

[3] C. Venard, E. Gagnier, Y.K. Lee, N. Leclaire, I. Duhamel, S. Evo, "Status of the experimental validation of the Frnech CRISTAL V1.1 package", Proc. of Int. Conf. on Nuclear Criticality and Safety, Saint-Petersburg, Russia, (2007).

OVERVIEW AND STATUS OF DOMESTIC AND INTERNATIONAL STANDARDS FOR NUCLEAR CRITICALITY SAFETY

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For many years, the United States (US) and international consensus standards for nuclear criticality safety (NCS) have provided guidance for those with hands-on operations involving fissionable materials. These consensus standards have been crucial to reducing the number of criticality accidents in process facilities. The last known criticality accident inside the United States was in 1978 (nearly 41 years ago) at the Idaho Chemical Processing Plant, and outside the United States, an accident occurred at Tokai-mura, Japan, in 1999 (20 years ago). The domestic consensus standards for NCS include the American Nuclear Society (ANS) standards. The ANS Standards Board, the NCS Consensus Committee, and the ANS-8 Subcommittee oversee the development and maintenance of these standards. Currently there are 18 standards in the ANS-8 series: 6 are in revision mode and 12 are in maintenance mode. The international consensus standards for NCS calculations, procedures, and practices are maintained and developed within the International Organization for Standardization, Technical Committee 85 on Nuclear Energy, Subcommittee 5 on Nuclear Fuel Technology, and Working Group 8, "Nuclear Criticality Safety." Eight standards are currently available, three standards are in revision mode, and four proposed standards are at various stages of development. This paper provides the NCS community with an overview and status report on domestic and international NCS consensus standards to stimulate interest and to support their continued development.

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GRS HANDBOOK ON CRITICALITY – NEW PUBLICATION IN 2019

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Since the early 1970ties the "GRS Handbook on Criticality" is a standard reference for criticality data. It is widely used by German regulators, and research institutes that are active in the field of nuclear safety analysis. In this work we present the extensions and enhancements of the latest edition of volume II of the "GRS Handbook of Criticality", published in 2019.

Volume II contains criticality curves (e.g. critical spherical volume and mass, critical diameter of an infinitely elongated cylinder, critical thickness of an infinitely large flat slab, infinite multiplication factor, ...) of a vast selection of fissile materials present in the nuclear fuel cycle. The criticality data is presented for various enrichments. For some materials also different moderators, neutron poisons or varying chemical configurations are presented. Part I of volume II contains $^{235}U/^{238}U$ systems, part II contains systems with Pu, ^{233}U and higher actinides.

Compared to the previous publication, the 2019 edition contains many updated criticality curves using the latest version of SCALE 6.2 [1]. The calculations were also extended by additional enrichments. All legacy data, available so far only in printed form, was digitized and is illustrated in the same standardized plotting format as the new data.

By means of one exemplary material the newly calculated criticality curves are presented and compared to legacy data.

A subset of the newly calculated data is validated by analyses of an extensive set of critical benchmark experiments, taken from the ICSBEP handbook [2]. These analyses contain sensitivity, uncertainty and correlation analyses on uncertain technical parameters and nuclear data.

[1] B.T. Rearden and M.A. Jessee, Editors, "SCALE Code System", ORNL/TM-2005/39, Version 6.2, Oak Ridge National Laboratory, Oak Ridge, TN (2017).

[2] OECD-NEA,"International handbook of evaluated criticality safety benchmark experiments," Tech. Rep. NEA/NSC/DOC(95)03, NEA Nuclear Science Committee, 2016

CURRENT STATUS OF NUCLEAR REGULATION IN JAPAN -FOCUSING ON NUCLEAR CRITICALITY SAFETY

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By reflecting the lessons learned from the accident of TEPCO's Fukushima-Daiichi Nuclear Power Plant which occurred on 11th March 2011, the new regulation body, Nuclear Regulation Authority (NRA), Japan has formulated the new regulatory requirements for the nuclear facilities. The feature of new regulatory requirements for the nuclear fuel facilities, such as fuel fabrication facilities and reprocessing facilities, are introduced in comparisons with the former requirements, focusing on the criticality safety issues. In addition, the present status of the safety review of nuclear fuel facilities under the new regulatory requirements is presented.

FEEDBACK FROM IAEA TRANSSC WORKING GROUP AND TECHNICAL EXPERT GROUP ON CRITICALITY

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One of IAEA's missions is to establish model regulations for the safe transport of radioactive material by all modes. The Transport Regulations (SSR-6) are important for all stakeholders including governments, regulators, operators of nuclear facilities, carriers, producers of radiation sources, cargo-handlers and the public. The mission is performed by the TRANsport Safety Standards Committee (TRANSSC). There are four TRANSSC Technical Expert Groups (TTEGs). The Criticality TTEG is one. The Criticality Working Group (CWG), established at TRANSSC n°34 in July 2017 as a general criticality safety WG, is the first WG under this TTEG. In the past, the CWG has met for two days, just before the TRANSSC meetings. This paper, together with a PATRAM 2019 paper, summarize some of the CWG exchanges:

Evolution of SSR-6 and the Advisory Material (SSG-26), e.g. resolution of inconsistency between SSR-6 and the International Maritime Dangerous Goods (IMDG) code for large freight containers, and conceptual solution for transport of empty, washed UF_6 cylinders.

Justification for SSR-6. In support of the IAEA Technical Basis (TB) efforts, the CWG has collected topics, suggested as being unclear or missing. Consensus has been reached e.g. on link between 10 cm minimum external package dimension and 10 cm cube entry, or is being reached, e.g. on high-speed air accidents and on definition/intent of confinement system. Remaining topics, even when perceived as understood, need documentation.

Interpretation of SSR-6, e.g. calculations for different (including very low) temperatures, exclusive use of large freight containers and conveyances, less severe test conditions than maximum. Usually involves the TB for the provisions or guidance.

A questionnaire to collect and share information on the TB, on current use and on requests for improvement of SSR-6. The responses to this questionnaire will be compiled for support of future use and development of SSR-6, SSG-26 and a TB.

14h00 - 15h40 > Track 5

IRSN APPROACH FOR CRITICALITY ACCIDENT ASSESSMENT

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The French Nuclear Authority for civil facilities (ASN) resolution n° 2014-DC-0462, published in 2014, presents the objectives to be achieved to prevent criticality accident in nuclear facilities (except reactor cores once loaded). Despite the provisions implemented to prevent such an accident, in line with the principle of defense in depth, the ASN resolution requires that licensees implement an emergency management to "limit the consequences of a criticality accident, in particular by implementing dedicated emergency management resources, when a conceivable combination of anomalies could lead to a criticality accident, and if they could provide significant benefits for the protection of people or the environment". The same approach is applied by French Nuclear Safety Authority for Defense-related facilities and activities (ASND).

This paper presents the main points assessed by IRSN (French institute for radiological protection and nuclear safety) to answer authorities' requests regarding the licensees' propositions to limit criticality accident radiological consequences. This assessment covers the following issues:

- detection of criticality accident (need for CAAS, probes implementation, maintenance and failure, detection without CAAS, etc.),
- emergency response (evacuation, assembly station, etc.),
- strategy to stop the criticality accident.

Each subject is addressed in the form of questions to ensure that the main issues are assessed. The main issues will be illustrated by examples drawn from previous IRSN assessment. Finally, the paper will present the latest works done by IRSN to support French nuclear authorities in case of criticality accident.

THE NEW VERSION OF THE CRITICALITY SAFETY GUIDE SHEETS COLLECTION

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The purpose of "The criticality safety guide sheets" is to gather useful elements in order to get sufficient knowledge to lead a criticality safety analysis for a nuclear installation containing fissile material. The first version of this guide was published in 2001 for the IQC/QCE (Ingénieurs Qualifiés en Criticité / Qualified Criticality Engineers) of the French CEA. It was composed of 27 sheets, each of them dealing with a specific topic such as bounding fissile medium definition, criticality control by limitation of the fissile material mass and neutron reflection.

This paper presents the new version of these criticality safety guide sheets, published in 2018. This new version is an enhancement of the former one and includes new elements, in particular sheets dealing with:

- principles of a criticality safety analysis,
- basics of criticality safety calculations,
- usual pitfalls in criticality safety.

Many new elements have also been included in order to update the guide:

- the new French criticality safety resolution,
- updated results of criticality calculations, performed with the CRISTAL V2 criticality safety package,
- many application examples...

The new criticality safety guide sheets are now provided during the training sessions for the new IQC/QCE organized by the INSTN (National Institute for Nuclear Science and Technology).

Furthermore, this new version is now available to all in French, including outside the CEA.

Some of the sheets are currently being translated into English and examples will be provided.

USE OF ANSI/ANS 8.6 STANDARD FOR CRITICALITY SAFETY APPLICATIONS IN THE MODERN WORLD OF ADVANCED SIMULATION CAPABILITIES

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Utilization of the standard ANSI/ANS-8.6-1983 "Safety in Conducting Subcritical Neutron Measurements in Situ" is discussed as a means for putting together a criticality safety evaluation for a fissile process where some information about the process is not known and/or when sparse computational validation data exists to achieve the necessary safety margins. Past and present ideas for utilizing the standard are given. A subset of the operational guidance contained in ANSI/ANS-8.6-1983 is discussed. A generic example for performing an in situ subcritical neutron multiplication measurement is given. The generic example illustrates multiple methods for analyzing the data with the intent of making the criticality safety case for attaining the required safety margin for the process.

EXTENSIVE STUDY OF THE HETEROGENEOUS REPARTITION OF THE MODERATION WHEN BOTH THE FISSILE MASS AND THE MODERATION ARE CONTROLLED

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This article presents an extensive study of the calculations performed in the configuration where the criticality safety is achieved by both controlling the mass of fissile material and the moderation (for example water) of a single unit. This case often occurs when the control of the fissile mass alone is not sufficient to economically or practically operate a process. This method is often used for the fuel fabrication where an important quantity of powder need to be handled but may also be met in other nuclear facilities and transportation. In this context, from a calculation point of view, a homogeneous repartition of the moderation within the fissile material is generally not a penalizing configuration. So a heterogeneous repartition of the moderation is then considered: it currently consists in a given part of the fissile material uniformly moderated by the entire quantity of the moderator, this system being surrounded by the rest of the dry fissile material and eventually another reflector (water, concrete, lead, etc.) [1]. This paper will firstly briefly discuss how to calculate safety limits for this kind of configuration, in the past and nowadays, using state-of-the-art algorithms. Then, new results will be presented for this kind of configuration with various enrichments and densities, some of them being more penalizing than those previously presented [1].

[1] V. Rouyer et al., "Updated rules for mass limitation in nuclear plants," ICNC 2003, October 20-24, Tokai-Mura, Japan (2003).

STATUS OF THE NEA INTERNATIONAL ACTIVITIES ON NUCLEAR CRITICALITY SAFETY

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More than 20 years have passed since the establishment of the Working Party on Nuclear Criticality Safety (WPNCS), in the Organisation for Economic Cooperation and Development (OECD)/ Nuclear Energy Agency (NEA). This paper reports on the recent activities of WPNCS and the associated bodies.

AN OVERVIEW OF THE UNITED STATES DEPARTMENT OF ENERGY'S NUCLEAR CRITICALITY SAFETY PROGRAM AND FUTURE CHALLENGES

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The United States (US) Department of Energy (DOE)/National Nuclear Security Administration (NNSA) Nuclear Criticality Safety Program (NCSP) was formally established in January 1998 as part of the Defense Nuclear Facility Safety Board (DNFSB) Recommendation 97-2 implementation plan. The NCSP was initially focused on seven NCSP tasks: Critical Experiments, Benchmarking, Analytical Methods, Nuclear Data, Training and Qualification, Information Preservation and Dissemination, and Applicable Ranges of Bounding Curves and Data. The first NCSP 5-year plan was published in August 1999. Now, more than 20 years later, NCSP continues to serve the United States and the nuclear criticality safety (NCS) community around five technical program elements (TPEs): Analytical Methods, Information Preservation and Dissemination, Integral Experiments, Nuclear Data, and Training and Education. NCSP has three chartered support groups, the Criticality Safety Support Group, the Nuclear Data Advisory Group, and the DOE Criticality Safety Coordinating Team, all of which provide technical advisement to the NCS community, to DOE, and to the NCSP manager. NCSP has a comprehensive mission and vision document that provides technical and budget priorities for TPE attributes and goals to ensure that the needs of the NCS community and DOE NCS are prioritized appropriately to meet the goals defined by the 1998 DNFSB recommendation. Like any DOE/NNSA program, NCSP faces significant challenges in the future that involves program funding, rising nuclear facility costs, aging critical assembly infrastructure, knowledge retention, and other issues. This paper discusses NCSP and provides some insights into these and other challenges moving into the future.

FUTURE CHALLENGES IN RE-ESTABLISHING A SOLUTION CRITICAL CAPABILITY IN THE UNITED STATES

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With the closure and de-inventory of the Los Alamos Critical Experiments Facility (LACEF) starting in 2004, the United States lost its only remaining solution critical assembly capability. Since then, the United States (US) Department of Energy (DOE) Nuclear Criticality Safety Program (NCSP) has identified investigating solution reactor design and location as a high priority goal in its Mission and Vision document. A joint US-France project called MIDAS was subsequently established to design, construct, and operate a new, state of the art, solution critical facility in France. This project was unfortunately cancelled in 2012 during the early conceptual design phase.

In 2018, Lawrence Livermore National Laboratory (LLNL) commenced work on the feasibility and conceptual design of a solution critical assembly for thermal and epithermal studies (SOCRATES) at LLNL leveraging existing nuclear facilities. Technical requirements for the new facility include creation, characterization, transfer, use, and disposal of a wide variety of actinide solutions leveraging existing capabilities of LLNL's Plutonium Facility. A versatile criticality facility would necessarily need to be co-located at the same site of the fuel fabrication facility (e.g., the Pu Facility) as off-site shipment of fissile solution is not permitted in the US. The mission of the solution critical facility would include: (a) studying the criticality characteristics of

both "old" and "new" actinide wet chemistry processes (nitrates, chlorides, sulfates, fluorides) with varying fuel concentrations; (b) allowing for a variety of geometric configurations; (c) allowing for operations in burst mode, "free run," and steady state; (d) investigating multiphysics dynamic characteristics of critical solutions; (e) providing a source of neutron and gamma radiation (and shields) for criticality alarm and dosimetry testing; and (f) providing activation and fission products for radiochemistry experiments.

This paper will summarize existing assets and future challenges for siting a solution critical facility at LLNL. Existing assets include an operational plutonium facility with ongoing enhancements for modern small-scale solution processing and enduring state of the art actinide and radiochemistry capabilities. The principal technical challenge is re-establishing the critical facility itself, which LLNL is well-positioned to do having previously operated a general purpose critical facility (Building 110), research and steady-state reactors (e.g., PLUTO and LPTR) and several prompt supercritical burst machines (e.g., FRAN, KUKLA, SUPER-KUKLA, BREN). However, there exist unique challenges of siting the facility at LLNL. For example, while Building 110 still exists, its safety and authorization basis is less than a hazard category 3 nuclear facility. Future utilization of this facility would require upgrading the safety basis to at least nuclear hazard category 2, which is a major change in mission and scope.

NUCLEAR CRITICALITY SAFETY BEYOND 2019

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To address future challenges to Nuclear Criticality Safety, it is prudent to revisit the past. In 1997, there were no smartphones, MCNP released version 4b, MONK 7 was in use, and 3.5 inch floppies were vogue. Even, so, the Defense Nuclear Facilities Safety Board was compelled to make Recommendation 97-2 [1] wherein a key point was made: "...when faced with the need to determine what must be done to avoid a chain reaction, they [criticality safety engineers] most frequently fall back on complex multidimensional Monte Carlo calculations. Their use of simplified methods and their reliance on published data are minimal."

Given the many-fold increase in computational capability, resources, and availability coupled with a generation of engineers that have grown up with smart phones, the need to ensure criticality safety engineers have a background in nuclear physics on a fundamental level including familiarity with assemblies near the critical state is more important than ever. This paper discusses the challenges Nuclear Criticality Safety Programs face in developing and maintaining criticality safety engineers with the requisite backgrounds and experience.

[1] Conway, John T., "Defense Nuclear Facilities Safety Board [Recommendation 97-2] Continuation of Criticality Safety at Defense Nuclear Facilities in the Department of Energy (DOE) Complex," Federal Register, Vol. 62 No. 103, pp.29118-29120 (1997).

Session 2 > Room C-D

9h00 - 10h40 > Track 3

IMPACT OF COVARIANCES BETWEEN CRITICALITY BENCHMARK EXPERIMENTS ON LICENSING

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We study the impact of correlations between benchmark experiment neutron multiplication factors k_{eff} on the validation of criticality safety assessments for two typical examples of a LWR fuel configuration: a fuel pool loaded with PWR fuel assemblies and a critical experiment taken from the ICSBEP handbook. For this purpose, the correlations between the experimental k_{eff} values are varied, and for each variation the validation procedure is carried out. The analysis is based on the evaluation of 57 benchmark experiments from three experimental series. Two different statistical validation procedures are compared, the first employing the Monte Carlo-Bayes procedure MOCABA, the second being based on a linear regression model allowing for covariances between different observations. The outcomes of the analysis suggest that traditional validation procedures without consideration of correlations between benchmark experiment k_{eff} values still lead to a sufficiently bounding treatment of the computational k_{eff} bias and its uncertainty. This should, however, be confirmed by running more test cases similar to the ones presented in this paper. Varying experimental k_{eff} correlations in criticality safety validation appears to be a less burdensome alternative to estimating these correlations by propagating uncertainties of a large number of experimental parameters to k_{eff} uncertainties. In fact, the variation procedure appears to be the only way to cover the correlation effect in the usual case where insufficient input information is available for quantifying the experimental k_{eff} correlations.

CORRELATION OF HST-001 DUE TO UNCERTAIN TECHNICAL PARAMETERS – COMPARISON OF RESULTS FROM DICE, SAMPLER AND SUNCISTT

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In this work we present a detailed uncertainty, correlation and sensitivity study of k_{eff} values with focus on uncertain technical parameters of ten experiments of critical high enriched uranium solutions with a thermal neutron spectrum. The experiments are documented in the ICSBEP as HEU-SOL-THERM-001(HST-001) [1]. The total Monte Carlo approach was chosen to allow all uncertain quantities to be sampled at once following their individual distribution functions. The stochastic dependencies between variables of different experiments were chosen based on available data.

The analyses were done individually and independent by GRS using the code SUnCISTT [2] and ORNL using the SCALE sequence SAMPLER [3]. Both Monte Carlo approaches rely on the neutron transport code KENO from SCALE 6.2.2 [4]. This enables a direct comparison of the implemented total Monte-Carlo methods.

Each of the two codes were used to analyze two different sets of uncertainties: The first evaluation is based on the uncertainties given in chapter 1"Detailed Description" of the HST-001 evaluation in the ICSBEP handbook. The second is based on the evaluated uncertainties given in chapter 2.0 "Evaluation of experimental data".

The uncertainty, sensitivity studies, and Pearson´s correlation coefficients for the k_{eff} values calculated are presented. A comparison with the correlation coefficients given in DICE [5] is discussed.

[1] OECD-NEA,"International handbook of evaluated criticality safety benchmark experiments," Tech. Rep. NEA/NSC/DOC(95)03, NEA Nuclear Science Committee, 2016.

[2] M. Behler, M. Bock, F. Rowold, M. Stuke, "SUnCISTT - A Generic Code Interface for Uncertainty and Sensitivity Analysis", Probabilistic Safety Assessment and Management PSAM12, Honolulu, Hawaii, USA, 22-27 June, (2014).

[3] B. T. Rearden, K. J. Duggan and F. Havluj, "Quantification of Uncertainties and Correlations in Criticality Experiments with SCALE," ANS Nuclear Criticality Safety Division Topical Meetings (NCSD2013), Wilmington, NC, 2013.

[4] B.T. Rearden and M.A. Jessee, Editors, "SCALE Code System", ORNL/TM-2005/39, Version 6.2, Oak Ridge National Laboratory, Oak Ridge, TN (2017).

[5] OECD-NEA: "Database for the International Criticality Safety Benchmark Evaluation Project (DICE)", September 2015 Edition, build 2.7, 2015.

THE INFLUENCE OF CHANGES IN NUCLEAR COVARIANCE DATA ON THE CALCULATION OF C_K FOR HIGHLY ENRICHED URANIUM SOLUTION SYSTEMS

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Since their initial development nearly 20 years ago, sensitivity/uncertainty (S/U) techniques have been increasingly applied to criticality safety validation. These techniques can generally be applied quickly and for a range of purposes after the initial investment of calculating sensitivities for a safety analysis model and a range of potential benchmark experiments. This paper discusses these applications and how they have been affected by changes in covariance data libraries.

Critical experiment selection is the most prominent area in which S/U methods impact validation efforts. These methods allow for rigorous calculation of similarity between safety analysis models and potential benchmarks. The most widely used of these methods is the integral parameter c_k, although other integral parameters such as E and some non-integral parameters have also been explored. The c_k parameter calculates the fraction of nuclear data–induced uncertainty shared between an application and an experiment; this metric is predicated on the assessment that nuclear data errors are the most likely source of computational bias. Therefore, values of c_k are inherently dependent on the nuclear covariance library used to calculate them and inevitably change when these data change.

Past studies have shown how much the c_k values can vary, especially in difficult validation cases like burnup credit. These significant changes are the result of differences in the relative uncertainties of nuclides with large sensitivities, and they imply differences in benchmark applicability to the safety analysis model. This paper demonstrates that more common criticality safety models involving single fissile species have significantly less variability of c_k between covariance libraries, thus enhancing confidence in the application of S/U methods for experiment selection in these scenarios. This paper presents case studies examining the impact of these changes on some hypothetical safety analysis systems.

UACSA PHASE IV: ROLE OF INTEGRAL EXPERIMENT COVARIANCE DATA FOR CRITICALITY SAFETY VALIDATION SUMMARY OF SELECTED RESULTS

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The subject of UACSA Phase IV was the quantification of covariances between neutron multiplication factors k_{eff} of criticality safety benchmark experiments due to uncertainties of system parameters shared by different experiments and the investigation of the impact of these covariances on criticality safety validation. Generally, these covariances have an impact on the computation of the k_{eff} bias and its uncertainty and, hence, on the best estimate plus uncertainty k_{eff} prediction for a given application case. Phase IV was divided into two sub-phases named as Phase IVa and Phase IVb. Phase IVa was based on an analytic toy model while Phase IVb was based on a set of critical experiments from the ICSBEP handbook. In both sub-phases, the task was to calculate the covariances and apply them to estimate the bias-corrected k_{eff} values of given application cases. In this paper we focus on the results for the bias-corrected k_{eff} values. A comprehensive summary of all exercise results will be presented in the final report on Phase IV.

11h10 - 12h50 > Track 3

ASSESSMENT OF NORMALITY FOR CRITICALITY SAFETY BIAS AND BIAS UNCERTAINTY CALCULATION

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The single-sided lower tolerance factors frequently used for non-trending assessment of the validation bias and bias uncertainty are sensitive to departures from normality. When used properly, the tolerance limits ensure that an appropriate fraction of the true population of applicable critical experiments lies above the calculated lower tolerance limit with the required statistical confidence level. One condition necessary to ensure that the appropriate proportion of the true population of k_{eff} values in the validation suite lies above the lower tolerance limit is that the assumption that the normality of the underlying population of critical experiments is valid or conservative.

This paper discusses various methods used to assess whether the assumption that the validation suite may be treated as a random sample drawn from a normal distribution is acceptable. Techniques for assessing the validity of this underlying assumption include common omnibus hypothesis tests for normality, assessment of sample skewness and kurtosis of the validation suite, and graphical techniques. These techniques are used to assess the nature and potential conservatism/nonconservatism imparted by various departures from normality where possible. A review of hypothesis testing is also presented to frame the discussion of omnibus normality tests. Additionally, two cases are analysed with these techniques to provide an example of how they should be implemented.

COMPARING THE WHISPER VALIDATION LIBRARY WITH MACHINE LEARNING METHODS

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This work summarizes and expands upon the current state of the art criticality validation methodology currently employed in the Whisper software distributed with the MCNP6.2 code. We show that the benchmark selection process can be sped up significantly by applying affinity propagation clustering to preselect benchmarks that are most likely be similar to an

application of interest. While for criticality safety a conservative estimate of bias is a must, we show that we can predict the expected bias using random forests when using k-eigenvalue sensitivities as features. Finally, we show that the nuclear data adjustment process can be performed in an informed manner by using the feature importance measures from the random forest. This work focuses on comparing and contrasting the various tasks done in Whisper with alternative methodologies used in data science.

A PROPORTIONATE APPROACH TO E_{PD}

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Dounreay Site Restoration Ltd. (DSRL) has historically taken a proportionate approach to dealing with the potential intristic bias in Monte Carlo calculations (E_{PD}), which is as follows:

'Any potential underestimation of k_{eff} from the intrinsic bias associated with the Monte Carlo code and neutron cross-section data library will be offset by the overestimation of k_{eff} due to the selection of conservative modelling parameters used in criticality safety assessments.'

This paper quantifies E_{PD} and the minimum overestimation of k_{eff} from typical conservative modelling parameters for systems of relevance to DSRL. Justification for DSRL's historic approach is made by comparison of E_{PD} and the minimum overestimation effect systems of relevance to DSRL.

MONTE CARLO UNCERTAINTY ANALYSIS METHOD IN "GADOLINIUM CREDIT" APPLICATIONS TO BWR CASK CONFIGURATIONS

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Implementation of Monte Carlo (MC) uncertainty analysis procedures in "Gadolinium Credit" criticality safety analysis of high-enriched BWR fuel loadings of CASTOR® V/52 casks is described. Validation of depletion calculations is based on evaluation of several radiochemical assay data sets. Maximum Likelihood and Bayesian data analysis methods including treatment modalities for incomplete multivariate data are employed to gain bounding Isotopic Correction Factors. Validation of criticality calculations is based on calculational evaluation of several critical experiments. Correlations between the resulting neutron multiplication factors due to the uncertainties in the applied nuclear data and due to the fact, that same material components (e.g., fuel rods) are used in different experiments, are estimated by means of MC methods.

14h00 - 15h40 > Track 10

RENEWAL OF IRSN TRAINING IN NUCLEAR CRITICALITY SAFETY

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In order to maintain competence or to train new employees, it is necessary to prepare and organize some training sessions suitable to the concerned audience. As a TSO (Technical and Scientific Support Organization), IRSN is particularly implicated in this issue, specifically in the field of human radiation protection, protection of the environment and nuclear safety.

IRSN created recently an "In-house University" containing different "Schools": "Assessment", "Emergency Planning/Response".... The main objective is to support IRSN employees following a specific route. Concerning the "Assessment" school, this route is first composed of a training on general topics (e.g. "how to perform an assessment?"), then of other trainings on specific topics ("nuclear criticality safety", "fire risks", "containement"...). Implemented teaching methods are innovative and less academic than it used to be. These methods have been taught by training professionals specialized in andragogy (methods and principles used in adult education). In this context, the "nuclear criticality safety" (NCS) training has been completely renewed by the SNC department (Neutronics and Criticality Safety Department) of IRSN. Scheduled over a period of one week, each day an expert teacher stays with the participants and interacts with the other speakers planned in the agenda, which allows dynamic discussions. Moreover, various exercises in groups (using paper boards, post-its, videos ...) give the opportunity to test the newly acquired knowledge. The session begins with general notions concerning NCS (physical properties, consequences of a nuclear criticality accident ...). Then, each control mode (mass, geometry ...) is discussed considering lots of examples from actual facilities. Finally, a review of past nuclear criticality accidents is done, and the impact of other risks (fire, flood, ...) on NCS is evaluated. A future project is to propose an international training in English intended in particular for other TSOs via the ENSTTI organization (European Nuclear Safety Training and Tutoring Institute).

MAINTAINING NCS CAPABILITY, CAPACITY AND COMPETENCE AFTER ENORMOUS ATTRITION

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SCOPE

As criticality safety is a niche profession, the training of new members has usually followed an apprenticeship pattern. Experienced engineers mentor incoming engineers through a variety of on-the-job training activities until the mentee demonstrates competence. Demonstration of competence has ranged from the simple decree of a manager to more structured programs that include more rigorous methods such as testing on policies and procedures, completion of technical assignments, and an oral board defense. However, if there is significant attrition within an organization, such as what the Los Alamos National Laboratory Criticality Safety Program suffered around 2012, mentorship of new staff limits the rate at which the organization can grow. If great care is not taken during the growth period, the mentorship experience can be diluted resulting in the compromise of the standards of training, and may result in additional attrition. A key attribute for success is to maintain the high standards of training for the trainee, complimented with adequate support from one or more mentors. The central conflict is the determination of priorities for mentors who are Qualified engineers with technical work assignments against their mentorship role, which is necessary to grow new capability, capacity and competence into the organization.

RESULTS AND CONCLUSIONS

Mentorship of new criticality safety staff is essential to grow and maintain any organization. Each engineer must balance work and mentoring priorities, with an emphasis on mentoring because new staff represents the future of the organization. Important characteristics of the mentor is an attitude of being willing and interested in mentoring, technical credibility based on qualification, experience or expertise, and enough soft skills to be an effective communicator and teacher. One of our more successful ventures involved leveraging people who have previous experience as criticality safety engineers or have a special skill set in one of the core technical competencies. A detailed and well-documented training program will optimize the efforts of the mentor; however, it does not change the fact that the critical element of the training program is the need for face-to-face interaction between mentors and mentees on a regular basis. One of the best ways to support this interaction is by co-location of mentors and mentees in the same workspace, including routine on-the-job mentoring visits within the operating facility. In conclusion, mentors must be fluent in both analytical and interpersonal skills, and there is no substitute for strong mentors in rebuilding a criticality safety program. With competent mentors, regular interaction, and meaningful work, mentees will build trust in mentors and the organization, which has been the basis for the recovery of qualified staffing levels for the Los Alamos National Laboratory NCS program.

CURRENT STATUS OF THE DOE/NNSA NUCLEAR CRITICALITY SAFETY PROGRAM HANDS-ON CRITICALITY SAFETY TRAINING COURSES

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In 2011, the U.S. Department of Energy/National Nuclear Security Administration (DOE/NNSA) Nuclear Criticality Safety Program (NCSP) developed and piloted a two-week Nuclear Criticality Safety (NCS) Practitioner course to support the training and qualification of new NCS staff. The course was developed in accordance with the American National Standard Institute/ American Nuclear Society (ANSI/ANS) standard for NCS training and qualifications (ANSI/ANS-8.26-2007). In 2013, an NCS Manager's course was developed for process supervisors, managers, regulators, and other professionals with NCS-related responsibilities. In 2017, an additional course was proposed for Criticality Safety Officers (CSOs). The baseline content for a CSO course is being defined by the NCSP Criticality Safety Support Group prior to training materials being developed. A pilot CSO course is planned for 2020. These courses consist of the following training components: classRoom education, facility training, and hands-on subcritical and critical experiments training. The 2-week Practitioner and 1-week manager courses are currently offered twice per year. The CSO course will be likely be offered once per year. The two-week Practitioner course offers a week of classRoom training, with practical workshops and exercises focused on teaching students how to perform an NCS evaluation. The second week of training involves hands-on critical and subcritical experiments and measurements. The first week is offered in Las Vegas, Nevada, at the DOE Nevada Field Office or the National Atomic Testing Museum. Depending on the student's clearance level, the second week is offered at Sandia National Laboratory (SNL) (uncleared and L-cleared students) or at the National Criticality Experiments Research Center (NCERC) (Q-cleared students). The one-week Manager's course is offered at SNL or NCERC, depending on clearance or interest, and includes classRoom and hands-on critical and subcritical experiments and measurements. The CSO course will likely be shorter than the manager courses, 3 days perhaps, and the lectures and hands-on content will graded for those supporting operations and criticality safety staff within a site NCS program. This paper provides an overview and status report for the DOE/NNSA NCSP training courses in NCS and to provide information about future course offerings.

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UNIVERSITY PIPELINE PROGRAM FOR THE EDUCATION OF FUTURE NUCLEAR CRITICALITY SAFETY PROFESSIONALS

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SCOPE

The Nuclear Criticality Safety (NCS) Division at Los Alamos National Laboratory (LANL) has established partnerships with Texas A&M University (TAMU) and the University of California, Berkeley (UCB) in order to develop a Nuclear Criticality Safety University Pipeline Program. Cooperation with other schools has also provided positive results, without a formal agreement. The goal of this program is to teach students the basics of NCS, preparing them to enter the field upon graduation. While Los Alamos has been the primary benefactor thus far, such a program benefits the entire U.S. nuclear industry by educating and training future employees in the basics of NCS.

The specifics of each school's program vary slightly, but each consists of a course that teaches the theory and application of NCS principles. The course outlines are developed with LANL NCS input and are taught with LANL assistance. The Y-12 and Lawrence Livermore National Laboratory (LLNL) NCS Divisions also participate in teaching the TAMU and UCB courses, respectively. The goal of LANL NCS pipeline is to pilot this program on a national scale to benefit an entire complex of new scientists and engineers, in order to combat the effects that attrition may have on the NCSP as a whole.

RESULTS AND CONCLUSIONS

The current Pipeline Program is set up such that students who perform well can participate in an internship in the LANL NCS Division the following summer, focusing on NCS practices and facility specific training for qualification. Standout students may proceed with a criticality safety oriented research project worked during the senior academic year. The culmination of the pipeline is the hiring of students as full time personnel. This practice reduces the uncertainty around knowing whether the individual: (1) has the technical skills, knowledge, and abilities to succeed in criticality safety, (2) will be able to effectively integrate within the organization, and (3) is interested enough in the discipline to reduce potential retention issues.

The program thus far has been a success. The TAMU course alone yielded 12 of 19 LANL NCS interns in the summer of 2018, and more than 40 students total were enrolled in the three programs during the fall 2018 semester. Improvements to the program are planned for the fall 2019 semester, including seeking to incorporate general qualification requirements into course syllabi. The second (spring) semester at TAMU provides students with continued interest in criticality safety an opportunity to enroll in a course focused on writing a full evaluation for a simulated process. Both LANL and Y12 are supporting this continuing education.

Currently in the works, LANL has committed to developing a Master's program focused in criticality safety. The goal is to streamline young professionals that have a strong knowledge of the science (or in some cases an art) involved.

The university pipeline results in several benefits: (1) reduced training time and costs, (2) interested students will naturally self-sort and pursue the discipline at the university level, and (3) a pipeline of criticality safety candidates is readily available within the DOE Complex so that unexpected organizational or mission changes can be reacted to with increased agility.

16h10 - 17h50 > Track 10

CRITICALITY SAFETY TRAINING AT CEA

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In October 2014, the French government adopted the Criticality Safety Resolution (2014-DC-0462) issued by the French Nuclear Safety Authority (ASN). This resolution was written after a 4-years period of work between different groups gathering licensees (CEA, EDF, ORANO – ex-AREVA...), experts, and the Technical Safety Organization (IRSN) [1] of the ASN. This resolution introduced a new regulatory text, replacing the prior, non-regulatory Fundamental Safety Rule I.3.c, issued in 1984.

In this new resolution, some articles are specifically dedicated to the organization and the training of the operators' nuclear criticality staff and/or nuclear workers. Specifically, the Criticality Safety Resolution confers four levels of training:

- Personnel working in areas where fissile material are manipulated,
- Nuclear workers manipulating fissile material,
- Criticality safety competent staff in charge of operations,
- Criticality safety expert engineers.

This paper presents how the Atomic Energy Commission (CEA) satisfies these regulatory requirements regarding its organization and training programs.

[1] S. Evo (IRSN), C. Manuel (ASN), "Status of French Regulations concerning Nuclear Criticality Safety", Proceeding of Int. Conf. on Nuclear Criticality and Safety (ICNC 2015), Charlotte, USA, Sept 13-17, 2015.

"CRITICALITY SAFETY ANALYSIS" TRAINING COURSE FOR ENGINEERS TO BE QUALIFIED IN CRITICALITY SAFETY

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The CEA's organization regarding criticality safety in its nuclear facilities is based on an operational line, backed up by support resources, and a control line. The operational line consists of a local organization in every nuclear installation containing fissile material; this organization is managed by one or several local criticality engineers named IQC/QCE (Ingénieurs Qualifiés en Criticité / Qualified Criticality Engineers).

The "initial training" for QCEs consists of two session courses organized by the INSTN (National Institute for Nuclear Science and Technology). The first session deals with principles of criticality safety whereas the second one provides "practical training". These training courses have been set up since 2001 and are open to CEA future QCEs but also to other engineers (CEA or not) who want to improve their criticality safety skills.

This paper details the content of the second session training course called "Criticality safety analysis". This session mainly consists of acquiring a practical know-how on writing a criticality safety analysis and the criticality safety calculation specifications in support of the criticality safety analysis. In order to achieve the latter, the session provides to the trainees the tools on reading, understanding and using a technical criticality calculation report; and last but not least the session also provides the minimum recommendations and good practices for writing the criticality safety procedures and requirements to be followed by the operators.

A CEA nuclear installation safety documentation serves as a guidance during the whole session. A technical tour of the installation is part of the training session.

Moreover, "The criticality safety guide sheets" are provided as guidelines during the session; these sheets are presented in another paper of this conference ("The new version of the criticality safety guide sheets collection").

CRITICALITY TRAINING FOR THE ACTIVE HANDLING FACILITY

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A training package has been developed and delivered to operators of the Active Handling Facility on the Sellafield Site. The purpose of the training was to improve their general criticality awareness, their understanding of the basics of criticality safety, and to understand the basis of the main criticality controls on the facility.

The training took the form of a half day interactive session for each group. The groups were small in order to encourage discussion amongst the attendees. Careful consideration needed to be given to the overall tone of the training to instil a healthy respect for the field of criticality safety without creating fear. Whilst this is an important aspect of any criticality training, it is particularly pertinent for this facility because it operated with a criticality incident detection system during a particular campaign to repackage some legacy fuel, but in recent years it has been operating under the terms of a criticality incident detection omission case.

At the same time as training the plant operators, the opportunity was taken to actively encourage them to consider any potential improvements that could be made to their Criticality Safety Case from an operational perspective. The process of actively seeking the opinions of operators led to mature discussions during the training. By actively following up their suggestions, this has helped to maintain a good trusting relationship. When operating staff see the impact of taking an active role in maintaining criticality safety, this then promotes a healthy safety culture. This paper explores this and the various other aspects that contributed to the overall success of the sessions.

TRAINING FOR FISSILE MATERIAL HANDLERS, SUPERVISORS, AND GENERAL PERSONNEL

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Hands on training for Fissile Material Handlers (FMH) from TA55 at Los Alamos National Laboratory (LANL) is being performed at Sandia National Laboratories and at the National Criticality Experiments Research Center (NCERC) at the Nevada Test Site. The courses are designed to give operators experience in taking special nuclear material critical in a controlled environment, and experience on how different parameters effect nuclear criticality for different types of systems. The Sandia course involves the use of an experimental reactor containing low enriched uranium fuel pins. Approach to critical experiments range from adding mass, adding water to an over massed system, separating two halves of the core, or removing fuel rods to achieve criticality. The last two experiments are designed to show the counterintuitive nature of nuclear criticality safety, which demonstrates precisely why we back away from any system that is a suspected process deviation. The experiments at NCERC are performed in conjunction with the critical experiments group at Los Alamos involving neutronically fast, reflected systems as well as an experiment involving polyethylene plates that mimics a solution system.

This curriculum also encompasses specific training for Criticality Safety Officers (CSO) as well as nuclear material supervisors,

known as Operations Responsible Supervisors (ORS) at LANL. As CSOs and ORSs interact more frequently than other individuals with the Nuclear Criticality Safety Division (NCSD), the training focuses on day to day interfacing with NCSD. These interfaces include how to request/schedule work with the division, how the Criticality Safety Evaluation (CSE) process develops, and the basics of a Credible Events Analysis (CEA) which determines normal and credible abnormal conditions within each operation. In addition, the class emphasizes the importance of an on-going and positive relationship between the NCSD and operations personnel.

Fissile material handling training is also offered in a graded scale at LANL, in order to give the necessary information to the correct people, depending on what that individual's work entails. For example, operators who are regularly handling nuclear material inside of a glovebox enclosure will receive very detailed criticality safety training, which includes examples of previous criticality accidents and the lessons learned from those events. However, an operator working with waste drums will receive less detailed instruction, as those activities are lower risk due to the nature of the process. This ensures that operators will be given pertinent information related to their specific process, thus aiding in the retention of the important objectives.

Session 3 > Room 1

9h00 - 10h40 > Track 1

DEVELOPMENT OF SUPERCRITICAL TRANSIENT MIK CODE AND ITS APPLICATION TO GODIVA CORE

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A Multi-region Integral Kinetic (MIK) code has been developed to analyze supercritical transients in fissile systems of arbitrary geometry and composition. This study tested the ability of the MIK code to control the computation cost and calculation accuracy by changing the update interval of the functions based on the feedback effects. Calculations were performed for the Godiva experiment. It was shown that the computation time could be reduced effectively without greatly reducing the calculation accuracy by chosing the appropriate interval.

EMPLOYMENT OF THE SINGLE EIGENVALUE MONTE CARLO METHOD TO SOME CRITICALITY SAFETY PROBLEMS; COMPARISON WITH A STANDARD DETERMINISTIC – MONTE CARLO APPROACH

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A Monte Carlo technique for calculating radiation responses inside and outside critical configurations is applied to a nuclear criticality safety problem and is compared to a standard approach employing a deterministic calculation of the adjoint flux that then defines the variance reduction parameters for a Monte Carlo run. The aspect of employing superhistories in the Monte Carlo technique is then examined on a second configuration based on Whitesides' k_{eff}-of-the-world problem.

THE HIGH-SPEED STATISTICAL CRITICALITY EVALUATION METHOD BASED ON THE MULTIDIMENSIONAL INTERPOLATION FOR ON-DEMAND CRITICALITY EVALUATION

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In the decommissioning of Fukushima Daiichi Nuclear Power Station (NPS), the removal of fuel debris cannot be avoided. The fuel debris is currently considered to be sub-critical. volume ratio of debris/water has strong sensitivity to the criticality. Thus, the criticality estimation is required in the debris removal process. While, the best estimation of the criticality is difficult due to the unclear properties and/or geometries of debris, therefore, random sampling with Monte Carlo calculation was used in current evaluation. However, the Monte Carlo calculation spend three days to one week for the evaluation due to its high calculation cost. Therefore, it is difficult that Monte Carlo apply to on-demand statistical criticality evaluation. For these background, multidimensional interpolation was applied to the statistical criticality evaluation. In this study, fundamental validation was examined by comparing with Monte Carlo results. The proposed method was 614,400 times faster than Monte

Carlo calculation, additionally, difference of mean value was 0.9 %dk. As a result, proposed multidimensional interpolation showed good agreement with direct Monte Carlo calculation. In the future work, simplification of the model, evaluation of applicable limit and operation procedure will be examined. This work is a part of "Development of Technologies for Controlling Fuel Debris Criticality" project supported by the Ministry of Economy, Trade and Industry (METI).

11h10 - 12h50 > Track 9

LESSONS LEARNED FROM THE ACCUMULATION OF URANIUM IN A GAS PURIFICATION SYSTEM

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On July 4, 2017, maintenance was being performed on an inert gas purification system used on a Research and Test Reactor (RTR) UAlx processing glovebox at the BWX Technologies, Inc. facility in Lynchburg, Virginia. The gas purification system has two large canisters filled with purification media. The media has two constituents; one removes moisture and the other removes oxygen from the inert gas. When the canisters were disconnected during maintenance, the operators saw what they believed was UAlx in the canisters. They stopped and notified Nuclear Criticality Safety (NCS) engineers.

Preliminary estimates of the mass were potentially in excess of 1 kg ^{235}U. The Emergency Operations Center was activated. More detailed Non-Destructive Analysis (NDA) measurements were later made and determined that the higher loaded container had less than 722 grams ^{235}U. This was less than the minimum critical mass in the containers. After 27 hours, the Emergency Operations organization stood down.

A Radiation Work Permit (RWP) was developed to guide removal of the material into individual ≤ 2.5 liter bottles. These bottles were then measured on a Material Control and Accountability (MC&A) qualified system and stored for disposal. An investigation was undertaken to determine how material accumulated in these containers, whether these systems were installed elsewhere in the facility, and if similar conditions existed in those systems.

Several similar systems were reviewed and evaluated by NDA. No accumulations were found. The investigation established three root causes and 18 corrective actions to prevent recurrence.

CRITICALITY SAFETY ASPECTS OF THE "BUMP LATCH" EVENT AT DUNGENESS B

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In 2009, an incident in the ex-reactor fuel route at Dungeness B (DNB) Power Station in the UK resulted in challenges to criticality safety controls. Subsequent assessments demonstrated that the geometry and amount of material involved could not have caused a criticality incident. This paper describes the incident and response and also subsequent work to strengthen the company's arrangements for criticality safety management.

DNB uses low-enriched UO_2 fuel, which is loaded into the reactor in the form of fuel assemblies. These are constructed on site. Each assembly contains a vertical stack of fuel elements, called a "stringer", which is attached to a fuel plug unit within a steel assembly tube. This process is known as "bump latching". The event occurred when foreign material prevented the correct bump latching of a stringer. This left the stringer hanging from the fuel plug unit with the risk that it might fall and severely damage the fuel. As an emergency measure, an unapproved moderating material (expanding polyurethane foam) was injected into the assembly tube below the stringer to minimise fuel damage in the event of a drop. This was done without consideration of criticality or fire risks.

The foam did not expand fully within the tube. Subsequently it was recognised that unexpanded foam could be an effective moderator, which might cause a criticality risk if the fuel dropped. The initial assessment inaccurately predicted that criticality might be possible.

There were no safety consequences from this event. The stringer was recovered without dropping it. However, this incident was rated as an INES level 2 event, due to degradation of defence in depth.

The introduction of foam contravened criticality controls and the potential criticality risk was not identified before it was used. Post-event investigations concluded that staff lacked adequate knowledge of criticality safety.

Following the event a wide range of criticality safety improvements were implemented within EDF Energy. These improvements included: the introduction of a formal Criticality Specialist role to provide on-site advice; production of a new company procedure for criticality safety management; update of criticality safety documentation; development and delivery of criticality safety training and strengthening of the capabilities of the central criticality team.

This event illustrates a number of key issues that should be considered by all organisations: How do you ensure that all personnel involved in activities that affect fissile materials are adequately trained in criticality safety? Are criticality safety controls obvious to everyone in the area? Is criticality safety managed effectively at your site?

CRITICALITY INCIDENT DETECTION DECISION MAKING: THE EVALUATION OF UNFORESEEN RISK

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A Criticality Accident Alarm System (CAAS) has an important role in minimizing the potential dose from a criticality accident should one occur in a process. The presence of one does not imply an intolerable or unacceptable risk of a criticality accident; it is there in order to mitigate the effects if an accident were to occur. Implicit with the presence of a CAAS is that for any process the risk of a criticality accident has been evaluated to be tolerably low but there is considered to still be a net benefit to the presence of an alarm system.

In the UK the decision is largely based on a test of either "reasonably foreseeable" after the loss of controls designated to maintain criticality safety, or based on consequences of dose below a certain threshold [1]. In the United States and many other countries the test is in accordance with ANSI/ANS standard 8.3 [2] where the quantity of fissile material present in a process triggers an evaluation of the CAAS need based on expert judgment. The decision is also made with consideration of the benefits and harms, as any system must provide a net benefit in terms of risk. This can be a difficult and subjective process. This process is made more difficult still in that part of the basis for CAAS is to provide mitigation for unforeseen events. It is intrinsically difficult to try to evaluate the unforeseen, and this has on occasion led to intractable contention that in turn has led to the installation of a CAAS based purely on the unforeseen component of risk. Installation may be entirely appropriate. Lacking any means of evaluating the unforeseen however, leaves the determination of the need for a CAAS open to challenge and potentially the wrong decision. Improvement to this situation could reap benefits of securing a (more likely) correct decision, clarified approach, and (in turn) time and cost savings.

Following the proposal developed in Reference [3] this paper sets out a method to estimate the significance of unforeseen criticality risk. The method can be used to assist in the decision to install a CAAS. It sets out a structure by which meaningful discussion may be had with stakeholders in order to arrive at a more informed decision for CAAS. The method employs a simple means of estimating the unforeseen risk contribution to the overall risk of criticality. By this means it may be determined whether the unforeseen risk is trivial, of concern (warranting further consideration) or significant (where a CAAS would likely be installed).

[1] K. J. Aspinall, J. T. Daniels, "Review of U.K.A.E.A. Criticality Detection and Alarm Systems 1963/64, Part 1: Provision and Design Principles", AHSB (S) R92 (1964).

[2] "Criticality Accident Alarm System", ANSI/ANS 8.3 (1997).

[3] N. Harris, "The Unforeseen Component of Risk when Considering the Need for a Criticality Accident Alarm System", Proceedings of the International Conference on Nuclear Criticality, Charlotte NC, USA, City & Country (2015).

CRITICALITY ACCIDENTS DETECTION AND MINIMUM ACCIDENT OF CONCERN: REVIEW AND DISCUSSIONS

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In order to detect the occurrence of a criticality accident, Criticality Accident Alarm Systems (CAAS) use the emission, at the beginning of the criticality accident, of an important flux of neutrons and gamma rays. CAAS are not a mean of prevention, but may limit the consequences of an ongoing criticality accident: even with a CAAS installed, personnel may die or be seriously irradiated. Once a criticality accident has just started, CAAS are intended to trigger an alarm for the evacuation of personnel in order to limit their doses.

In order to determine the best location of the CAAS probes in a facility, a "minimum" criticality accident to be detected should be defined. This "minimum" criticality accident is also named "minimum accident of concern" (MAC) in the standards related to CAAS. Some of them ([1] [2]) are currently under revision. The current MAC defined in these standards corresponds to a slow kinetic criticality accident for unshielded solution systems but its origin, its expression and its justification are not well documented and discussions about the MAC are in progress. This paper brings some technical points about these discussions, in addition to those already provided in a previous article [3].

In particular, this paper will remind the link between the detection and the radiological consequences of criticality accidents. Then the various expressions used to define the MAC over the years will be discussed and compared. In addition, the lessons learned from past criticality accidents, the use of past experiments (like CRAC divergences) and the possible use of computer tools will be discussed to better define the MAC. Finally, the specificities of the devices detecting criticality accidents below the MAC are discussed for the definition of a MAC value.

[1] International Organization for Standardization, "Nuclear energy – Performance and testing requirements for criticality detection and alarm systems," ISO 7753, 1st ed., (1987).

[2] "American National Standard Criticality Accident Alarm System," ANSI/ANS-8.3-1997, American Nuclear Society, (1997).

[3] M. Duluc, "Criticality accidents detection, minimum accident of concern and slow kinetic criticality excursions: experimental data and discussions for solution systems," Int. Conf. on Nuclear Criticality ICNC 2011, paper 6-03, Edinburgh, Scotland, (2011).

ASSESSMENT OF RE-CRITICALITY IN SEVERE ACCIDENT CONFIGURATIONS USING MCNP AND MELCOR

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The paper presents a method to evaluate the potential re-criticality issues in severe accident configurations. This method is developed by Tractebel ENGIE, using respectively MCNP and a dataset of MELCOR calculations to answer Severe Accident aspects of gaining importance issued from the WENRA RL, in particular the validation of IVMR "In-Vessel Melt Retention" concept and Fukushima type situations where non borated water is injected in the primary loop. The present work focuses on the in-vessel phase and more particularly on a TMI-2-like accidental configuration. Finally a "surrogate model" - regression model – is presented for the prediction of the multiplication factor for TMI-2-like configurations.

EXPERIENCE IN EVALUATIONS OF CRITICALITY IMMEDIATELY AFTER ACCIDENTS WITH THE DESTRUCTION AND MELTING OF NUCLEAR FUEL AT NPP

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Difficulty and importance of determining possibility of occurrence of criticality in the accidents with destruction and even melting of nuclear fuel at NPP differ significantly whether this determination occurs immediately or after a long period of time since accident. Immediately after the accident, when situation is not yet stabilized, there is no sufficient amount of data on the state of fuel and moderator and this requires additional efforts to assembly conservative computational models.

Current report reviews three evaluations of criticality that were carried out at the Kurchatov Institute immediately after accidents at the 4th block of the Chernobyl NPP in 1986, in the fuel assembly washing tank at the Paks NPP in 2003 and at the Fukushima NPP in 2011. The conclusions made after these evaluations did not differ significantly from the conclusions of subsequent analyzes performed after a long time after the accident. However, earlier criticality evaluations may influence the management of the accident progression and the determination of optimal strategy of emergency response. After the accident at the Paks NPP, results of computational evaluation of the criticality were taken into account when preserving the subcritical state of tank with destroyed fuel and when removing fragments of fuel assemblies from the tank. In emergency response during Chernobyl accident, such results were not taken into account. In the accident at the Fukushima NPP, there was no opportunity to inform emergency workers with results of computational evaluation of the criticality.

NUMERICAL ANALYSIS OF CRITICALITY OF FUEL DEBRIS FALLING IN WATER BY COUPLING COMPUTATIONAL FLUID DYNAMICS AND THE CONTINUOUS ENERGY MONTE CARLO CODE

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Accurate evaluation of the dynamic behavior of fuel debris in water in terms of criticality safety is an essential part of the decommissioning process. The purpose of this study was to demonstrate that it is possible to evaluate criticality safety using the actual dynamic behavior of fuel debris in water with a calculation system coupling DEM, MPS and MVP. Experiments and calculations of ball mills sedimentation in water were carried out first in order to verify the accuracy of MPS-DEM calculations. We found that in cases of high-density material, the sedimentation shape could be calculated; in cases of light-density material, however, the sedimentation shape could not be calculated accurately because of the difference in the spreading of the ball mills between the experiment and simulation. Changes in the effective multiplication factor at each time were then calculated using DEM, MPS and MVP. Results showed that it was possible to calculate the effective multiplication factor of fuel debris falling in water with this calculation system. We demonstrate that it is possible to evaluate criticality safety using the actual dynamic behaviour of fuel debris in water with a calculation system that includes DEM, MPS and MVP.

EXPLORATORY INVESTIGATION FOR ESTIMATION OF FUEL DEBRIS CRITICALITY RISK

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For the criticality safety of the operation treating the fuel debris in FUKUSHIMA Daiichi Nuclear Power Plant, the reactivity effect of its geometrical change has been investigated and the developed procedure has been applied to a trial analysis of a postulated scenario for the purpose of its verification.

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SUPERCRITICAL KINETIC ANALYSIS IN A SIMPLE FUEL DEBRIS SYSTEM BY MIK CODE

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To begin the removal of fuel debris at Fukushima Dai-ichi Nuclear Power Station, the consequences of possible criticality accidents must be evaluated in advance. Predictions of the region-dependent number of fissions can be especially important. Such evaluations are necessary for dose evaluations for workers, machines and the public. In this work, instead of traditional point kinetic analysis, space-dependent kinetic analysis was performed using the MIK code, which is a unique method based on Monte Carlo neutron transport calculations. The target of analysis was a simple spherical fuel system where fissile regions are coupled weakly in terms of neutronics. Specifically, we sought to reveal the relationship between the water level surrounding the fuel debris and the region-dependent number of fissions during criticality accidents, as it is thought that the water level surrounding the fuel may influence both the reactivity and the weakly coupling between fissile regions and have an impact on the region-dependent number of fissions. Through the observation of Cij function, which is a key parameter for space-dependent kinetic analysis using the MIK code, the characteristics of weakly coupled systems are revealed. In addition, the relationship between the water level and the region-dependent number of fissions at supercritical condition is evaluated and discussed.

MULTIPHYSICS COUPLING ANALYSIS FOR SPENT FUEL POOL LOSS OF COOLANT ACCIDENT

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The Fukushima accident has shown that improvements on the nuclear safety are more than key for the success of the nuclear industry and its public support. Accordingly, reinforced efforts should be directed to continue the improvement of the studies of reactor power systems and any other systems containing radiative materials. Among the latter are the spent fuel pools (SFP). In case of a Loss of Coolant Accident in a SFP, not only the spent fuel assemblies' integrity can be compromised due to its inadequate decay power cooling, but also in some hypothetical scenarios serious questions could arise about the maintaining of adequate criticality margins. These questions are important due to the risk of fission products release to the environment that may result from significant fuel assemblies' (FAs) damage. In the SFP the spent FAs are usually arranged in racks and immerged in borated water, which serves both as coolant and reactivity control. The distance between the racks is a key parameter in reactivity control. A hypothetical scenario leading to a reduced criticality margin in the SFP may occur after a significant water loss due to a crack on the SFP generated by a seism or an earthquake and inadequate make up water flow. In this postulate scenario, and depending on the size of the crack and the makeup water flow rate, the SFP level and the coolant characteristics (temperature, boron concentration) could evolve overtime leading eventually to reduce criticality margins. The main mechanisms explaining the possible SFP criticality margin reduction is related to the effects of the decrease of the water level in the gap between fuel racks, the effect of the water level and density inside the fuel assemblies and the boron concentration. These mechanisms will modified the neutron moderation and absorptions rate and the neutronics coupling between fuel racks. An adequate numerical model to study this accident requires therefore coupled neutronics and thermohydraulic calculations. The thermal-hydraulics model is necessary to correctly predict the water levels in the SFP and the racks and should include phenomena such as convective and conductive heat transfer, change from water to air natural convection, boiling and two-phase flow and hydrogen production and combustion from zirconium–steam reaction. The neutronics model should be able to accurate predict the SFP reactivity and the phenomena associated to an hypothetical criticality accident. In this paper, a tool using multiphysics coupling is presented to asses this type of accidents. This new tool has been developed using OpenFOAM code to solve the continuum mechanics equations (e.g. Thermal-hydraulics equations) and the Monte Carlo code Serpent for the neutronics aspects. Due to its geometric complexity, the Spent Fuel Pool (SFP) containing the racks and the FAs is simplified as a porous medium. A numerical case is presented to illustrate the capability of the tool to simulate different racks and FAs' configurations.

MULTIPHYSICS SIMULATION OF TWO CRITICALITY ACCIDENT EXCURSIONS IN LADY GODIVA USING MCATK

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In this paper, multiphysics simulations of the two Lady Godiva criticality accidents are performed by coupling the Monte Carlo Application ToolKit (MCATK) to a simple one-dimensional thermomechanics solver. Accurate simulation of past criticality accidents is necessary to estabilish capabilities for predicting and preventing future accidents. This work is a first attempt at validation of the MCATK multiphysics capability against real criticality accidents. We find that MCATK is able to predict the number of fissions for the two accidents to within roughly a factor of two. Given the model simplifications and the uncertainty of the information about accidents and their initial conditions, this agreement is thought to be quite good. These initial results indicate that MCATK could be a viable tool for criticality accident modeling, but that more validation is warranted.

CRITICALITY ACCIDENT SAFETY ANALYSIS: QUESTIONS AND PARTIAL ANSWERS PROVIDED BY DEDICATED EXPERIMENTS CONDUCTED ON CRAC AND SILENE

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Many studies and models in different fields concerning criticality accidents have already been carried out in different countries, notably based on experiments carried out in France on the CRAC and SILENE installations [1,2]. However, some of the more sensitive issues in safety analysis, such as the triggering of the criticality alarm and the management of the post-accident phase, are difficult to address. These questions are part of what could be called the "Criticality Accident Problem". Specific experiments described below have been conducted on SILENE to help provide information on the post-accident phase that would be encountered following the triggering of the criticality alarm in order to be better equipped to manage any urgent actions to be managed.

[1] F. Barbry et al., "Review of the CRAC and SILENE Criticality Accident Studies", Nucl. Sci. Eng., 161, p. 160 (2009).

[2] F. Barbry, "A review of the SILENE criticality excursions experiments", Proceedings of the Topical Meeting on Physics and Methods in Criticality Safety, Nashville, Tennessee, Sept. 19-23, 1993, American Nuclear Society (1993).

Session 4 > Conference Room LOUIS ARMAND

9h00 - 10h40 > Track 4

THE SANDIA CRITICAL EXPERIMENTS PROGRAM WHAT ARE WE DOING FOR YOU NOW?

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Recent activities in the critical experiment program at Sandia National Laboratories are described. The process by which critical experiments are done under the Nuclear Criticality Safety Program of the US Department of Energy is discussed. The experiments that are being actively pursued at Sandia National Laboratories are listed and briefly described. The most recent experiment series conducted in the facility is described and preliminary results of that experiment are presented.

NEUTRONIC DESIGN OF BASIC CORES OF THE NEW STACY

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Japan Atomic Energy Agency, JAEA, is conducting the renewal program of the STACY (Static Experiment Critical Facility), from a homogeneous solution system to a heterogeneous water moderated system, in order to verify the criticality calculation considering the fuel debris which have been produced in the accident of Fukushima Daiichi Nuclear Power Station. The first

criticality of the new STACY is scheduled at the beginning of 2021. After the first criticality, it is necessary to perform a series of critical experiments with basic experimental cores in order to gain a proficiency of operators and grasp the uncertainty that accompanies the result of critical experiments in STACY.

Prior to the construction of the new STACY, a series of neutronic calculations were carried out for licensing and planning first series of critical experiments.

In this paper, possible configuration of the basic experimental cores and their limitations are discussed and presented. Series of critical experiments with the basic experimental cores will be performed with the results of this study.

IMPROVEMENTS IN VOID REACTIVITY WORTH MEASUREMENTS USING A LOAD CELL AS PRESSURE SENSOR

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Benchmark quality critical systems containing lead (Pb) have been notoriously difficult to construct due, at least in part, to the malleability of lead. In 2016 and again in 2018, a system was constructed using highly enriched uranium and lead plates on the Comet critical assembly at the National Criticality Experiments Research Center (NCERC) for the purpose of providing a configuration suitable for a benchmark evaluation where lead voids could be introduced and their reactivity worth measured. This experiment followed the form of the Zeus experiments documented in several benchmarks.

Due to the malleability of the lead plates, the measured excess reactivity for these configurations was affected by how much pressure was applied when remotely closing the stack. Early measurements were inconsistent in the amount of pressure applied. After a load cell was installed on the Comet critical assembly, the corresponding pressures and measured data could be compared more reliably.

The results show that void reactivity worth is more accurately measured using this load cell and that results are more consistent with calculated values. In this system, reactivity was linear with pressure over the range of pressures that were measured. These configurations are part of a larger project which includes a heterogeneous low enriched uranium and lead system as well as a plutonium and lead system.

THERMAL EPITHERMAL EXPERIMENTS (TEX): TEST BED ASSEMBLIES FOR EFFICIENT GENERATION OF INTEGRAL BENHCMARKS

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The goals of the TEX Program are to address the recognized integral experiment needs of the US Department of Energy's Nuclear Criticality Safety Program (NCSP) by executing critical experiments with NCSP fissile assets that span a wide range of fission energy spectra, from thermal (below 0.625 eV), through the intermediate energy range (0.625 eV to 100 keV), to fast (above 100 keV). Three test bed assemblies have been designed: a ^{239}Pu assembly that uses Zero Power Physics Reactor (ZPPR) plutonium metal plates, a ^{235}U assembly that uses highly enriched uranium (HEU) metal Jemima plates, and a ^{233}U assembly that uses ^{233}U oxide ZPPR plates. All three test bed assemblies have the same basic design: layers of fuel interspersed with varying amounts of polyethylene moderator, which is used to tune the neutron fission spectrum, and a thin polyethylene reflector to reduce the effects of Room return. The assemblies were designed with as few materials as possible to provide clean benchmarks that would be useful in highlighting any nuclear data problems with the fissile materials. Additionally, the assemblies were designed to be easily modified to include high priority diluent materials of interest.

Five TEX-Pu baseline experiments, containing only Pu ZPPR plates and polyethylene moderators, were completed in 2018, as well as five complimentary experiments that tested tantalum as a diluent material in the assemblies. Tantalum is a material that is often used in high-temperature fissile material operations, but is under represented in the International Criticality Safety Benchmark Evaluation Project (ICSBEP). An evaluation is currently being prepared for inclusion in ICSBEP, and calculational results show some disagreement in the most intermediate energy cases, which worsens with the addition of tantalum. Additional cases are planned to test the absorption properties of chlorine, iron, and magnesium, thermal scattering laws for polyethylene and plexiglass, and a TEX-Pu variation with higher ^{240}Pu content.

Five TEX-HEU baseline experiments, containing only HEU metal Jemima plates and polyethylene moderators, have received final design approval and are scheduled to be assembled in the summer of 2019. An additional 11 experiments have been designed to examine the effect of a hafnium absorber on the TEX-HEU configurations, and those experiments are scheduled to be conducted in 2020. Additional planned diluent materials include lithium and chlorine. LLNL is also working on the design of an experiment to provide benchmark data for low temperature (-40 C) applications, based on the TEX-HEU design.

Final design has begun on the ^{233}U oxide baseline configurations. Initial calculations indicate that it is not possible to create a fast critical assembly using the available stock of ^{233}U oxide ZPPR plates, and so this series will only include thermal and intermediate cases. Final design is currently ongoing for these experiments.

TITANIUM AND ALUMINUM SLEEVE EXPERIMENTS IN WATER MODERATED 4.31% ENRICHED UO2 FUEL ELEMENT LATTICES

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Approach to critical experiments performed at the Burnup Credit Critical Experiment (BUCCX) at Sandia National Laboratories were utilized to measure of the effects of titanium and aluminum sleeves in the fuel array on the critical array size. The array was fully reflected by water with the approach parameter being the number of fuel elements. The inverse count rate as a function of the number of fuel elements was extrapolated to zero to obtain the critical array size for each configuration. The critical experiments complete a series of experiments with titanium, that provide the first integral test of newly-evaluated titanium nuclear data. The experiments are designed to provide criticality safety benchmarks with significant reactivity from titanium and equivalent configurations with aluminum sleeves.

Seventeen configurations were addressed by the critical experiments. The first of the experiments had no sleeves in the array and was intended to provide a baseline against which the experiments containing sleeves could be compared. Eight critical experiments had titanium sleeves in various quantities and arrangements near the center of the fuel array. Eight critical experiments had aluminum sleeves in the same numbers and arrangements as in the eight experiments containing titanium sleeves. The sleeves, which have an inner diameter approximately 1 cm larger than the outer diameter of the fuel elements, are each centered around a fuel element and positioned between the upper and lower grid plates of the assembly. The reactivity worth of the titanium sleeves ranged from 2 % to 10 % depending on the experimental configuration.

DESIGN METHODOLOGY FOR FUEL DEBRIS EXPERIMENT IN THE NEW STACY FACILITY

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The new criticality experiments facility STACY will be able to contribute to the validation of criticality calculations related to the fuel debris. The experimental core design is in progress in the frame of JAEA/IRSN collaboration. This paper presents the method applied to optimize the design of core configurations of the new STACY to measure the criticality characteristics of pseudo fuel debris focused on Molten Core Concrete Interaction (MCCI) debris. To ensure that a core configuration is relevant for code validation, it is important to evaluate the reactivity worth of the main isotopes and the k_{eff} sensitivity to their cross sections. If the sensitivities profiles are similar to those of the configuration to be validated, it is potentially feasible to provide relevant feedback on its nuclear data. In the case of MCCI debris described in this study, silicon is the nucleus that has the highest k_{eff} sensitivity in the concrete. Therefore, some parameters of the core configuration, as for example the lattice pitch or the core dimensions, were adjusted using optimization algorithm to research efficiently the optimal core configurations to obtain high sensitivity of silicon capture cross section. This method allows exploring a large space of possibilities by limiting the number of calculations. Two examples of designs tested using this approach are presented in this paper. The first study was performed on a simple square core configuration with ideal conditions. In the second study, the core was divided in two zones to investigate the interest of having both an experimental zone and a driver zone. Based on these results, realistic series of experiments for fuel debris in the new STACY could be defined to obtain an interesting feedback for the MCCI.

SOLUTION CRITICAL EXPERIMENTS PARTIALLY REFLECTED BY LUCITE

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A series of critical experiments performed at the Bettis Atomic Power Laboratory in the early 1950s on a highly enriched homogeneous solution assembly [1] is described. The assembly consisted of a 91.44 cm (36 inch) inner diameter and 182.88 cm (72 inch) high cylindrical stainless steel fissile solution tank surrounded by a 132.08 cm (52 inch) inner diameter and 177.8 cm (70 inch) high cylindrical water reflector tank. The fissile solution was partially reflected on the bottom by a 86.36 cm (34 inch) diameter and 25.4 cm (10 inch) high solid Lucite reflector. Bare and water reflected clean critical solution heights are provided for dilute highly enriched aqueous solutions of uranyl nitrate ($UO_2(NO_3)_2$) for eight concentrations with H/^{235}U ratios between 1633 to 1776. The results were experimentally corrected to 20 °C. A preliminary benchmark model is described and MC21 results are provided using ENDF/B-VII.1 and ENDF/B-VIII.0 cross sections. Hydrogen in the Lucite was modelled using both a free-gas treatment and the ENDF/B-VIII.0 hydrogen bound in Lucite thermal neutron scattering law (TSL) in order to assess the sensitivity to the Lucite thermal scattering treatment for this experiment.

[1] J. R. Brown, B. H. Noordhoff, W. O. Bateson, "Critical Experiments on a Highly Enriched Homogeneous Reactor," WAPD-128, Bettis Atomic Power Laboratory, May 1955.

WARM CRITICAL RUNS IN SUPPORT OF THE KILOPOWER REACTOR USING STIRLING TECHNOLOGY (KRUSTY) EXPERIMENT

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Three warm critical runs were conducted in support of the KRUSTY experiment. The purpose of the warm critical runs was to determine key parameters such as the reactivity temperature coefficient of the fuel and the heat transfer coefficients which are important to the modeling of the neutronic and thermal dynamic behavior of the KRUSTY experiments.

14h00 - 15h40 > Track 4

HISTORY AND FUTURE OF TEMPERATURE REACTIVITY EXPERIMENTS AT VR-1 REACTOR

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Calculation tools for fuel transport criticality safety have to be validated against comprehensive sets of experimental data. One of the aspects is the validation of code ability to predict the reactivity changes with temperature. Unfortunately, in the temperature range of interest for normal operations, i.e. between -40°C and +38°C, there is a shortage of available experimental data. At the VR-1 zero power reactor, as operated by the Czech Technical University in Prague, temperature reactivity experiments were established in 2011. These were originally for the purpose of education and training. Over time, the understanding of experimental conditions and uncertainties in VR-1 has been improved along with associated modeling tools and analytical methods. In 2017, there was an interest in the VR-1 experimental data from the UK, to support the validation of the MONK criticality safety code for temperature-dependent calculations. From this interest, further work has been carried out to support the applicability of the data for such a purpose and to make these data available to the criticality safety community. Also, from subsequent interactions, priorities for future deployment of temperature-reactivity experiments at VR-1 reactor were outlined to reduce the gaps in criticality safety temperature-dependent experimental data. This paper summarizes the history and present status of temperature-reactivity experiments at the VR-1 reactor and related calculation efforts. Finally, the outlook for proposed future experiments is given.

THE EFFECT OF TEMPERATURE ON THE NEUTRON MULTIPLICATION FACTOR FOR PWR FUEL ASSEMBLIES

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The effect of temperature on criticality safety evaluations is an area of significant international interest. This is because temperature may affect physical parameters, such as material density or phase, as well as neutronic parameters such as reaction cross-sections. For example, there has recently been work undertaken to determine the thermal scattering of H bound in ice for inclusion in nuclear data libraries.

In order to understand the variation associated with temperature dependent calculations for systems containing water, an inter code comparison benchmark has been undertaken through the WPNCS. The benchmark considers two cases:

- A PWR-type fuel assembly within a thick water reflector.
- An infinite array of PWR fuel assemblies.

The neutron multiplication factor is calculated at five different temperatures (233, 253, 293, 333 and 588 K), in order to test the variation in reaction cross-section with temperature as well as the change in thermal scattering data for bound H.

Each case has been handled in two parts: firstly a fresh fuel case was examined to consider the trend of variation in the calculated neutron multiplication factor with temperature; secondly, two irradiated fuel cases were considered to identify if these trends remain consistent as the burn-up of the fuel was increased.

This paper will report the preliminary results of this benchmark study, discussing the trends that are present and the variation based upon the data libraries and nuclear criticality codes used.

USE OF BWR COLD CRITICAL BENCHMARKS FOR CODE VALIDATION

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Teollisuuden Voima Oyj operates two BWR units at Olkiluoto. Cold critical measurements carried out at the reactor units can be used as benchmarks for the validation of codes for burnup credit. The use of the spent fuel critical benchmarks offers a straightforward means for the validation of the criticality safety analysis codes with an approach referred to as the combined validation approach. This approach was tested with CASMO-4E and MCNP which are the codes used at TVO for the Gd credit criticality safety analysis of BWR fuel storage facilities.

20 simplified benchmarks were prepared on the basis of the cold critical core geometry and the nodal operating histories of the fuel assemblies. The simplification procedure in the benchmark preparation was based on earlier work that has demonstrated that the effect on the system characteristics is almost negligible. SIMULATE-3 core follow-up calculations were used for the determination of the nodal operating histories of the fuel assemblies. The operating histories were used for the calculation of the fuel rod nodal isotopic composition of each fuel assembly with CASMO-4E. The calculated spent fuel compositions were applied in an effectively full core MCNP calculation of the critical benchmark.

In this work Monte Carlo codes MCNP5 and MCNP6 and cross section libraries based on JEF-2.2 and ENDF/B-VII.1 were applied in the criticality calculation. The results were compared to the core analysis software CASMO-4E/SIMULATE-3 and JEF-2.2 based cross section data. The comparison to SIMULATE-3 results show that the simplified benchmark description can be applied without significant impact on the results. The results were also compared as a function of the control rod withdrawal fraction around the critical control rod position. In general the results of MCNP and SIMULATE-3 were consistent.

The results show that the Monte Carlo calculation of the BWR cold critical benchmarks can be carried out with a reasonable effort. These benchmarks can improve the quality of the code validation and aid in reducing unnecessary conservatism in the criticality safety analysis. The variety of fuel designs and the range of burnups effectively covers the Gd credit application. The results indicate that BWR cold critical benchmarks can be used for the combined validation approach. On the basis of the validation it should also be possible to justify the validity of the 5 % delta-k-eff depletion uncertainty for Gd credit application.

STEADY-STATE BENCHMARK EVALUATION OF THE TREAT M2 AND M3 CALIBRATION EXPERIMENTS

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The Transient Reactor Test Facility (TREAT), located at Idaho National Laboratory (INL), is part of the Materials and Fuels Complex (MFC) and was originally designed to evaluate reactor fuels and structural materials. The TREAT reactor is a high enriched, air cooled, graphite moderated, and graphite reflected core. This thermal spectrum reactor is heavily influenced by the graphite material properties. The fuel is dispersed in a graphite matrix which provides a large thermal feedback safety mechanism as well as the ability to dissipate large amounts of heat from the fuel and into the graphite. These characteristics are necessary for the transient operations for which TREAT has been historically used. The M2 and M3 experiments, originally conducted in 1985, were designed to capture the transient impacts on EBR-II Mark-II fuel pins. Prior to the actual fuel pin irradiation, flux wire tests (calibration experiments) were conducted to quantify reactor conditions. Benchmark models of the TREAT M2/M3 system were developed to evaluate the reactor. Large uncertainties within the core arise from the fuel impurity composition and graphite thermal scattering which have been quantified using the developed models. Calculations were completed using the Monte Carlo code Serpent 2.1.29 with both the ENDF/B-VII.1 and ENDF/B-VIII.0 graphite libraries. The major difference between these two cross section libraries arises with the introduction of nuclear graphite thermal scattering libraries which directly impact the TREAT reactor physics. The ENDF/B-VIII.0 30% porous nuclear graphite library is a more accurate representation of the TREAT graphite matrix than the ENDF/B-VII.1 ideal graphite. Combined with impacts from both the boron and hydrogen impurity content within the fuel, the system uncertainty can be bound. At this stage, the resulting benchmark model is able to represent the TREAT system within 1% of the expected experimental eigenvalue. To further reduce the deviation from known reactor behavior, additional analysis with the benchmark models has been completed to reduce the possible range for the hydrogen impurity concentration. Addressing the hydrogen impurity content based on TREAT thermalization physics, the predicted eigenvalue and its overall uncertainty can be further improved to agree within 0.1% of the benchmark eigenvalue.

CRITICALITY TESTING OF RECENT MEASUREMENTS AT THE NATIONAL CRITICALITY EXPERIMENTS RESEARCH CENTER

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In regards to nuclear data, some applications may lack validation experiments which reduces confidence in predicted results. This work presents an approach to design new criticality experiments which have similar k_{eff} cross-section sensitivities to an application of interest. This process involves simulations to generate cross-section sensitivities to a parameter of interest (such as k_{eff}), a gap analysis to determine which existing benchmarks are most similar to the application, and an experiment optimization. This work compares measurements and simulations of recent experiments at the National Criticality Experiments Research Center (NCERC). In particular, results from the KRUSTY experiment are used as an example to demonstrate the approach presented in this work.

VALIDATION OF NEW SILICON EVALUATION IN SPECIAL CORE OF LR-0 REACTOR

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Silicon is an important part of the Earth's crust, thus its scattering properties can influence the safety of final repositories of spent nuclear fuel. A new evaluation of silicon cross-section is a part of the recent effort of the Oak Ridge National Laboratory and supports the International Energy Agency INDEN project as well. Additionally, the new silicon dioxide TSL matrix has been recently published in ENDF/B-VIII.0 data library. Due to this fact, the validation experiment of these evaluations was performed in the special core of the LR-0 reactor. Silicon dioxide sand was poured into a dry channel and placed into the centre of the core. Simulations of this experiment were performed using MCNP6 code and a sensitivity study was made in TSUNAMI-3D. The simulation was performed using several combinations of silicon and uranium cross-sections. The new TSL matrix for SiO_2 was also tested. Comparison of experimental results with calculations shows that new evaluations together with the new TSL matrix give results closer to reality.

BENCHMARK EVALUATION OF SAXTON PLUTONIUM PROGRAM UO_2-FUELED CRITICAL LATTICES

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In 1965, a series of water-moderated critical experiments using UO_2- and UO_2/PuO_2-fueled lattices was performed at the Westinghouse Reactor Evaluation Center in Waltz Mill, Pennsylvania. These experiments were performed to verify the nuclear design of the Saxton partial plutonium core via experiments varying in arrangement, fuel type, lattice pitch, and insertion of additional simulated reactor materials. Criticality, buckling, power distribution, reactivity, control rod worth, soluble poison, and power peaking measurements were performed. The current benchmark evaluation effort includes critical experiments of seven single-region lattices of stainless-steel-clad UO_2 fuel rods (5.74 wt.% ^{235}U) in Room temperature water of various lattices and a couple with variations in material placement within the core center. Analyses were performed with MCNP6 and ENDF/B-VII.1 to evaluate experimental uncertainties and biases with benchmark model development. Dominant uncertainties include fuel rod pitch, manganese content of the cladding, outer diameter (thickness) of the cladding, and fuel rod diameter. Calculations of the eigenvalues are within 1s of the benchmark values, except for Case 6, which is within 2s. Future work includes evaluation of the remaining critical configurations, as well as the additional reactor physics measurements performed in this experimental series.

INVESTIGATION OF THE IMPACT OF THE PREDICTION ACCURACY OF THE BURN-UP CODE SYSTEM SWAT4.0 ON NEUTRONICS CALCULATION

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Criticality safety of the fuel debris from the Fukushima Daiichi Nuclear Power Plant is one of the most important issues, and the adoption of burnup credit is desired for criticality safety evaluation. To adopt the burnup credit, validation of the burnup calculation codes is required. JAEA developed a burnup calculation code SWAT4.0 to obtain reference calculation results of the isotopic composition of the used nuclear fuel. In this study, assay data of the used nuclear fuel irradiated by the Fukushima Daini Nuclear Power Plant Unit 1 (2F1ZN2 and 2F1ZN3) and Unit 2 (2F2DN23) are evaluated to validate the SWAT4.0 code for solving the BWR fuel burnup problem. The calculation results revealed that the number densities of many heavy nuclides and fission products show good agreement with the experimental data. To investigate the applicability of SWAT4.0 to the criticality safety evaluation of fuel debris, we also evaluated the effect of isotopic composition difference on k_{eff}. The differences in the number densities of ^{235}U, ^{239}Pu, ^{241}Pu, and ^{149}Sm have large impact on k_{eff}, whereas the other heavy nuclides and fission products have negligible impacts on k_{eff}. The reactivity uncertainty related to the burnup analysis was less than 3%. Therefore, we could adopt the maximum permissible multiplication factor, i.e., upper safety limit of k_{eff} of 0.95, which is the conventional criterion of the subcriticality assessment. These results indicate that SWAT4.0 appropriately analyses the isotopic composition of BWR fuel, and it has sufficient accuracy to be adopted in the burnup credit evaluation of fuel debris.

Session 1 > ❷ Room A-B

9h00 - 10h40 > Track 11

PROGRESS OF CRITICALITY CONTROL STUDY ON FUEL DEBRIS BY JAPAN ATOMIC ENERGY AGENCY TO SUPPORT SECRETARIAT OF NUCLEAR REGULATION AUTHORITY

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Criticality control of the fuel debris in the Fukushima Daiichi Nuclear Power Station would be a risk-informed control to mitigate consequences of criticality events, instead of a deterministic control to prevent such events. The Nuclear Regulation Authority of Japan has administrated a research and development program to tackle this challenge since 2014. The Nuclear Safety Research Center of Japan Atomic Energy Agency, commissioned by the authority, is conducting activities such as computations of criticality characteristics of the fuel debris, development of a criticality analysis code, preparation of criticality experiments, and development of a criticality risk analysis method.

NUCLEAR CRITICALITY SAFETY IMPACTS OF ADDITIVE MANUFACTURING

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The 2016 opening of the Additive Manufacturing User Lab (AMUL) at the Y-12 National Security Complex (NSC) has brought about innovative solutions to nuclear criticality safety problems. However, the use of this new technology is not without its challenges. Items made using additive manufacturing, also known as "3-D printing," can have different material properties than conventional manufacturing and add additional nuclear criticality safety concerns such as reflection and interstitial moderation. Appropriate engineering judgement is necessary to determine the best applications of additive manufacturing in processing fissile materials.

CRITICALITY CHARACTERISTICS OF FUEL DEBRIS MIXED BY FUELS WITH DIFFERENT BURNUPS BASED ON FUEL LOADING PATTERN

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The fuel debris produced by the accident of the Fukushima Daiichi Nuclear Power Station (1F) is probably in a state of mixture of burned fuels with different burnups each other. In such a case, the mixing ratio of burned fuels in fuel debris would affect its criticality. This report shows computation results of criticality characteristics of fuel-debris compositions prepared by mixing nuclide compositions of burned fuels in various patterns based on a fuel loading pattern. The results indicate that fuel debris is potentially subcritical when 1-cycle fuels, whose average burnup is several GWd/t, are included homogeneously in fuel debris because remaining ^{155}Gd and ^{157}Gd in 1-cycle fuels works to reduce neutron multiplication. The results also indicate that ^{155,157}Gd/^{235}U ratio well characterize criticality of fuel debris.

APPLICATION OF THE NEUTRONIC PART OF THE NUCLEAR SIMULATION CHAIN OF GRS TO ACCIDENT TOLERANT FUEL SYSTEMS – FIRST RESULTS

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The nuclear simulation chain of GRS [1,2,3,4] provides powerful tools in the field of nuclear in-core and ex-core safety analysis. As for neutron physics, this toolbox includes i.a. code compilations related to criticality, cross-section processing, static and transient reactor core analysis, and nuclide inventory determination e.g. for waste management applications. It is well established and validated for light water reactor systems featuring square or hexagonal-shaped fuel assemblies, with low-enriched uranium dioxide fuel and zirconium alloy based cladding material (UO_2/Zr) comprising the main constituents.

In terms of improved robustness under severe accident conditions, recently diverse innovative fuel materials are under investigation within the superordinate concept called "Accident Tolerant Fuels" (ATF) [5]. Here different materials amend or replace the well-known fuel components, introducing new isotopes to the system such as iron, chromium, aluminum, silicon, carbon, or others. Besides the thermo-mechanical properties primarily under scope, some of these materials feature neutronic properties which differ from the standard UO_2/Zr system and thus impact on various safety-related nuclear in-core and ex-core characteristics. In addition to new materials, also different arrangements of materials occur, e.g. multi-layered cladding materials or heterogeneous fuel forms as TRISO-like particles in an inert matrix forming water-moderated fuel rods [5]. These arrangements pose new challenges to accurate cross-section processing in terms of resonance self-shielding, resonance treatment, and collapsing to few-group cross-sections. As for the overall neutronic calculation chain, this directly affects the problem-dependent reactor core simulations as well as waste management applications.

The robust and reliable applicability of the overall GRS nuclear simulation chain to ATF needs to be demonstrated. First results of this ongoing work are shown here, featuring pin-cell and assembly-wise criticality, cross-section processing, and burn-up calculations using different sequences from the SCALE code system with ENDF/B-VII.1 based cross-section libraries [6]. Comparative analyses using the SERPENT [7] code with ENDF/B-VII.0 based continuous energy cross-sections for criticality calculations, few-group cross-section generation for reactor physics applications, and inventory determination are performed. First code-to-code and experimental benchmark exercises are discussed. Reasonable or good agreement between the different calculation systems as well as to the benchmark experiments have been found so far. Hence, up to now no insurmountable obstacles have been observed to accurately account for the peculiarities of some ATF as compared to the standard system. However, for a final evaluation, further investigations need to be performed and assessed.

[1] A. Schaffrath, A. Wielenberg, M. Sonnenkalb, R. Kilger, "The nuclear simulation chain of GRS and its improvements for new ALWR and SMR typical phenomena", Proceedings of the Intl. Conf. 12th International Topical Meeting on Nuclear Reactor Thermal-Hydraulics, Operation and Safety (NUTHOS-12), Qingdao, China, October 14-18, 2018, (2018).

[2] Zilly, M.; Bousquet, J.; Velkov, K. et al., "PWR cycle analysis with the GRS core simulator KMACS", AMNT 2018 - 49th Annual Meeting on Nuclear Technology, 29-30 May 2018, ESTREL Convention Center Berlin.

[3] Zilly, M., Bousquet, J., Pautz, A., "Multi-Cycle Depletion with the GRS core simulator KMACS: BEAVRS Cycles 1 and 2", (to be published at) M&C 2019 - International Conference on Mathematics and Computational Methods Applied to Nuclear Science and Engineering, 25-29 August 2019, Portland, Oregon, USA.

[4] Bostelmann, F., Krzykacz-Hausmann, B., Aures, A., Zwermann, W., Velkov, K., "Sensitivity Indices for Nuclear Data Uncertainty Analysis", Proceedings of BEPU 2018 – ANS Best Estimate Plus Uncertainty International Conference, Real Collegio, Lucca, Italy, 13-19 May 2018.

[5] S.J. Zinkle, K.A. Terrani, J.C. Gehin. L.J. Ott, L.L. Snead, "Accident tolerant fuels for LWRs: A perspective", Journal of Nuclear Materials, 448 (2014) 374-379.

[6] B. T. Rearden and M. A. Jessee, Eds., SCALE Code System, ORNL/TM-2005/39, Version 6.2.3, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2018). Available from Radiation Safety Information Computational Center as CCC-834.

[7] J. Leppänen et al., "The Serpent Monte Carlo code: Status, development and applications in 2013," Ann. Nucl. Energy, 82 (2015) 142-150.

Session 2 > Room C-D

9h00 - 10h40 > Track 10

NUCLEAR CRITICALITY SAFETY TRAINING AT THE NATIONAL CRITICALITY EXPERIMENTS RESEARCH CENTER

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Nuclear criticality safety grew out of the ranks of experimentalists studying the physics of chain-reacting systems at critical experiment facilities. Consequently, critical experiment facilities provide the best forum for conducting training in nuclear criticality safety. The sole remaining facility in the United States capable of conducting general-purpose nuclear materials handling including the construction and operation of high-multiplication assemblies, delayed critical assemblies, and prompt critical assemblies is the National Criticality Experiments Research Center (NCERC). The Los Alamos National Laboratory Advanced Nuclear Technology Group, consisting of approximately 50 full-time personnel representing an unparalleled national resource of expertise, supports NCERC. Hands-On Nuclear Criticality Safety Training is conducted at NCERC for fissionable material operators, supervisors, managers, and criticality safety analysts. This paper discusses the curriculum and benefits of hands-on training conducted at NCERC.

EDUCATION AND TRAINING AT VR-1 REACTOR FACILITY. CAN BE BENEFITING FOR CRITICALITY SAFETY ENGINEERS?

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Hands-on training with critical or subcritical assemblies is recommended in standards as a part of training and qualification for criticality safety engineers. It is supported by argumentation that it can provide better understanding of the factors that contribute to criticality safety. To develop the idea the paper shows the example of the VR-1 training reactor operated by the Czech Technical University in Prague, and describes its education and training capabilities and activities. It relates them to the perspective of criticality safety.

CRITICALITY AUGMENTED REALITY TRAINING AID

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Quite often, operators (or those outside the criticality safety community) ask why there isn't a single limit that they can adhere to, and simply guarantee criticality safety. Unfortunately, whilst there are values that guarantee criticality safety, these tend to be of little operational benefit and unrealistic as candidate 'Operating Rules'.

The real issue to convey, in a meaningful fashion, are the principles behind the required set of controls and how the interplay between multiple parameters can allow larger masses etc. that can be demonstrated to remain safely sub-critical.

Of crucial importance, therefore, is presenting the information in such a way so as it can be understood by on-plant staff and not just criticality safety practitioners who are experienced in the output of traditional safety assessments.

Cerberus Nuclear have developed their virtual reality environment, utilised for radiation shielding and characterisation applications, into a virtual reality platform to overlay, in real time, the criticality parameters associated with the system being considered within the environment. Our intent is to introduce the user into this environment in order to enhance their understanding of criticality safety in an intuitive way, enabling them to see – in real time – the effect their actions are having on a system's reactivity.

The first stage of this development is a simple two-body problem assessing parameters: mass, fissile concentration, reflector thickness and separation. The user will be able to hold a simulant fissile item and approach a second, with the reactivity (k-effective) displayed. The mass and fissile concentration in each of the items can also be varied, as well as reflection of the entire system, again to show the effect on reactivity.

By monitoring the reactivity under normal, and fault, conditions it is then made clear to the user that simple changes have a large effect on the reactivity, and thereby on criticality safety, assessment of which must demonstrate sub-criticality for all manner of scenarios.

Underpinning this virtual reality environment is a series of Monte Carlo (MCNP) calculations. In order to make the environment as realistic and fluid as possible, fast and accurate interpolation between data points is required. This is provided by a basic machine learning algorithm.

Progress to date is discussed, in terms of underpinning MCNP models, the virtual reality environment(s), and development of the machine learning algorithm. In addition, short and long-term goals of the project are presented.

SAFETY ANALYSIS REPORT FOR PACKAGING SHIELDING AND NUCLEAR CRITICALITY SAFETY COURSES DEVELOPED AND CONDUCTED BY OAK RIDGE NATIONAL LABORATORY

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The US Department of Energy (DOE) Packaging Certification Program, Office of Packaging and Transportation, is offering Safety Analysis Report for Packaging (SARP) Shielding and Nuclear Criticality Safety (NCS) courses for SARP generalists and analysts. The SARP generalist course is designed for project managers, supervisors, NCS/shielding subject matter experts (SME), and SMEs in non-NCS/shielding technical areas (e.g., structural, thermal, package design) who need to improve their understanding of how NCS/shielding analyses fit into the broader spectrum of SARP documentation. The generalist course provides an overview of the regulations and guidelines for the criticality and shielding analyses included in a SARP, and it illustrates how the NCS/shielding chapters integrate with the other parts of the SARP. After course material is presented, students in the generalist course will review an actual SARP document to examine key elements of the shielding and criticality analyses. The analyst course provides detailed instructions on the radioactive material package shielding analyses and NCS evaluation fundamentals needed by analysts/practitioners, including safety analysts and/or technical reviewers, to prepare and/or review technical analyses for SARP documentation. The analyst course also provides an overview of regulations and guidelines, in addition to detailed in-class exercises associated with package shielding and NCS analyses. Analysis teams will be faced with staged SARP examples in which several important decision processes in the generation of a SARP will be demonstrated and discussed. Both courses are offered at Oak Ridge National Laboratory, and for interested students, the analyst course is offered for one hour of graduate credit (course ME-690) from the University of Nevada at Reno. An overview of the course will be provided to ensure that the NCS community is aware of the course and its intent to support the shielding and NCS aspects of packaging and transportation analyses for SARP activities.

CRITICALITY ACCIDENT PHENOMENOLOGY: NUMERICAL EXPERIMENTS AS A LEARNING TOOL

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For decades, the training of French criticality experts has implied performing critical experiments on dedicated devices, which have all been shut down during the last few years. A work to substitute for these reactors as a learning tool has then been undertaken, which is presented here.

The developed tool calculates the evolution of a super-critical solution of uranyl nitrate placed in a tank, by solving point kinetics equation and taking into account every meaningful physical phenomenon (radiolysis, thermodynamics, heat exchange, dilatation...) in a simplified way. Reactivity feedbacks are implicitly implemented; they are estimated from neutronic pre-calculations done with APOLLO2 [1] for a given set of physical properties (void fraction, concentration, temperature). Users can configure experiments, tune physics and display physical observable variation with time through a graphical user interface.

This paper presents a detailed description of the tool, some elements about its implementation in the classRoom, as well as a few lessons learned and perspectives.

[1] R. Sanchez, J. Mondot, Z. Stankowski, A. Cossic and I. Zmijarevic, "APOLLO 2: a user-oriented, portable, modular code for multigroup transport assembly calculations", Nuclear Science and Engineering, 100 (3), pp. 352-362, https://doi.org/10.13182/NSE88-3 (1998).

REVIEW OF IRSN WORK REGARDING NUCLEAR CRITICALITY ACCIDENT

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This article presents a review of the French Institute for Radiological Protection and Nuclear Safety (IRSN) work regarding nuclear criticality accidents during the last 15 years. After the separation in 2002 of the French Atomic Energy Commission (CEA) and IPSN (the latest becoming IRSN), the IRSN nuclear criticality safety department had to develop its own skills and knowledge about criticality accidents.

This article will present the background and the history of the research, with the legacy of the work performed by the CEA/IPSN team in Valduc. After reminding the framework for this field, dictated by the IRSN objectives and constraints, the strategy and the various achievements accomplished by IRSN in this field will be presented. In particular, the article will cover in detail the following "phases" that can be retrospectively drawn from this global work. Initially the appropriation and the analysis of past data and tools were necessary. Then the time for the first developments, the first experiments and the first collaborations came. Next, the close connection with the dosimetry field became apparent.

Finally, the perspectives and the needs for this field will be presented. It will be emphasized the need for the use of experimental facilities that can study this hazard and the need for a strong national and international collaboration.

IMPACT OF CRITICALITY ACCIDENT CHARACTERISTICS ON SELLAFIELD CRITICALITY EMERGENCY ARRANGEMENTS

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The UKs nuclear site licencing requirements are non-prescriptive. In terms of accident prevention, fundamentally, they simply require the overall risk of worker harm from the operations to be as low as reasonably practicable, (ALARP) [1]. Further, they only demand that 'adequate' emergency arrangements are in place to deal with a nuclear accident. There is thus no requirement in law or national standards to have to assume any particular criticality accident characteristics. There is only a requirement to robustly justify the assumptions that are used.

Deployment of 'upfront' emergency countermeasures to mitigate the worker dose effects of a potential criticality accident are typically very cost intensive, with the benefits of the investment (hopefully) never realised, other than perhaps worker reassurance. These include capital costs such as provision of passive bulk concrete shielding, or an active criticality accident detection and alarm system. They also result in facility lifetime operational costs for periodic maintenance, testing and refurbishment. The assumed characteristics of the criticality accident, for example number of fissions, or power dynamics, or a certain level of accident source self-shielding, have an obvious and often significant impact upon these countermeasures including necessary thicknesses of radiation shielding and extent of immediate evacuation zones.

Historically, standard bounding assumptions and numerical accident characteristic values have been deployed at Sellafield to underpin its criticality emergency plans. However, in recent times there has been a growing need to re-examine how these

values are used and their potential level of conservatism in order to address a number of pressing challenges. Going forward these challenges are expected to increase as the site moves its focus from fixed scope nuclear fuel reprocessing operations to large-scale site remediation and accelerated radiological hazard reduction. This necessitates application of the concepts of programme and holistic ALARP, where criticality risk has to be balanced with all other risks to seek an optimised safety justification across many differing and sometimes competing priorities and hazards across the site. On the horizon are a desire to re-utilise land currently occupied by evacuation zones and site new fissile handling facilities adjacent to longstanding worker populations to deal with a severely constrained site. In addition, security drivers to deal with the modern day terrorist threat that conflict with longstanding criticality escape arrangements are here now.

This paper describes some of the challenges in criticality emergency plan development at Sellafield and how they are being addressed now and looking forward into the future. It will highlight that a lack of experimental or computational data for criticality accidents involving moderated powders or slurries has the potential to lead to non-optimum solutions.

[1] UK Health and Safety at Work Act, 1974.

NEEDS AND STATE OF THE ART IN CRITICALITY DOSIMETRY AND DOSE RECONSTRUCTION TECHNIQUES FOR MEDICAL MANAGEMENT OF CRITICALITY ACCIDENT'S CASUALTIES

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The medical management of victims of a radiological accident is often driven by the information on the dose distribution or dose at organs at risk that is the main pertinent information expected. Since the Chernobyl accident with the feedback experience on the medical management of highly exposed liquidators, there is nowadays a medical management to treat patients if possible before clinical signs appear and therefore to develop a treatment strategy based in particular on dosimetry information. For criticality accidents, dosimetry is more complex, because of the possible high doses, high dose rates and complex gamma/neutron fields [1].

A high dose from a criticality accident requires dose estimation with a short delay to be effective. It is important to segregate the different contributions of the radiation field, due to the difference in biological detriment. As the neutron dose is mainly deposited in the first few cm of the body that implies to take into account the morphology specificity, difference in organ doses could be up to 30%.

This article presents the various technics used (physical retrospective dosimetry, cytogenetic, activation of blood and hairs and nails, Monte Carlo simulation, etc.) to estimate doses in case of criticality accident. Then, the needs for this specific field of dosimetry will be presented, including firstly the necessity for an international collaboration and cooperation, in order to maintain and to share the few facilities still available, and secondly to have scientific cooperation for future developments and improvements.

[1] International Atomic Energy Agency, Nuclear Accident Dosimetry Systems. Proc. Panel Vienna, 1969, IAEA, Vienna (1970).

Session 4 > Conference Room LOUIS ARMAND

9h00 - 10h40 > Track 4

ANALYSIS OF THE CRITICALITY BENCHMARK EXPERIMENTS UTILIZING UO_2F_2 AQUEOUS SOLUTION IN SPHERICAL GEOMETRY

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The analysis of the criticality experiments using an intermediate enriched UO_2F_2 aqueous solution in spherical geometry is presented. An evaluation of the total experimental uncertainty of k_{eff} was performed within the framework of the International Criticality Benchmark Evaluation Project (ICSBEP). The total experimental uncertainty was evaluated to be σ_{tot}=0.0065. The largest contribution to the total experimental uncertainty is due to the uncertainty in fuel enrichment (σ=0.0055). In addition, the uncertainty in k_{eff} due to the uncertainty in temperature and nuclear data is evaluated and presented in this paper. Through the evaluation process it was found that the multiplication factor is highly sensitive to the variations in the solution temperature. Special care was put into analysing the temperature dependence of the thermal scattering kernel. The component of the temperature coefficient of reactivity due to the thermal scattering law was determined to be: -14.28 $\pm$ 0.10 pcm/K.

The total temperature coefficient of reactivity accounting for Doppler broadening, thermal scattering law, and water density effects was calculated to be: -24.4 ± 1.7 pcm/K. In addition we estimated the uncertainties due to nuclear data. This was done by using SUSD3D and SANDY codes. It was estimated that the uncertainties in nuclear data contribute approximately 400 pcm to 1200 pcm to the uncertainty in the calculated k_{eff} depending on the covariance data library used. The evaluated criticality safety experiment was published in the ICSBEP Handbook under the identifier IEU-SOL-THERM-005.

CRITICALITY ANALYSIS OF NCA CRITICAL EXPERIMENTS SIMULATING SFP UNDER LOW MODERATOR DENSITY CONDITIONS

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Nuclear critical assembly (NCA) experiments simulating fuel rods and fuel storage racks under low moderator density conditions in the spent fuel pool (SFP) were conducted to obtain the criticality characteristic of the SFP during loss-of-coolant accidents. In these experiments, a 9 × 9 boiling water reactor (BWR) fuel assembly loaded in an aluminum, stainless steel, or borated stainless steel fuel storage rack was simulated and criticality experimental data such as critical water levels were obtained. It was realized that the aluminum storage rack cases had high reactivity against water densities in the test region due to neutron coupling effect. Then, the effective neutron multiplication factor k_{eff} values were calculated via the criticality safety assessment code MVP3.0 using libraries generated from JENDL-4.0 and ENDF/B-VII.1. The discrepancies between the measure d and calculated k_{eff} values for all cases with various types of storage racks under various moderator densities were found to be within 250 pcm. In addition, there were some differences in k_{eff} between the two libraries possibly due to different ^{1}H in H_2O thermal scattering models: the k_{eff} values using ENDF/B-VII.1 showed positive biases of about 200 pcm in all test cases

DETAILED DESIGN OF AN EPITHERMAL/INTERMEDIATE CRITICAL EXPERIMENT USING THE SANDIA NATIONAL LABORATORIES CRITICAL FACILITY

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Nuclear criticality safety evaluations commonly take credit for neutron-absorbing materials. The normal practice for validating the calculational process to develop safety limits and margins of safety is to compare the safety case models with calculations of various critical experiments of a similar composition. Because there is a lack of critical experiments with various relevant neutron-absorbing materials mixed with fissile material, a new approach is being developed. This new approach uses small amounts of these credited materials in a critical assembly and assesses their worth and neutron energy spectral characteristics using absorption reaction rates. This approach is proposed as a new method for nuclear data validation. The assessment will involve measured central reactivity worth in a critical or near-critical assembly and subsequent comparison with calculated values from Monte Carlo computer analyses. Additionally, four configurations are proposed which replace fuel rods in the driver region with absorber rods. These configurations will allow for examination of the effects of the material in a more thermalized spectrum experiment selection in these scenarios. This paper presents some case studies examining the impact of these changes on some hypothetical safety analysis systems.

This paper documents the detailed design portion of the critical experiment design process, focusing on development of a capability to test epithermal/ intermediate energy cross sections for materials using the 7uPCX critical experiment facility. Tantalum is specified as the test material to be examined.

RESULTS OF A NEWLY EXPANDED COG CRITICALITY VALIDATION SUITE

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The COG suite of criticality benchmarks has been formally expanded from 591 to 2,256 to cover the entire energy range from thermal to fast neutron spectra under a variety of reflector and moderator conditions and fissile materials. COG results have been compared with benchmark values from the International Criticality Safety Benchmark Evaluation Project Handbook for ENDF/B-VIII.0 and JEFF-3.3. COG results have been also compared with available MCNP results from the IAEA 'Trkov' validation suite. Most of the results agree with the benchmark values within ±3σ. About 13% of the total cases are outside this ±3σ range. Sources of error may come from 1) cross section data, 2) possible errors from modeling of the experiments, or 3) benchmark experimental or evaluated data itself. Good agreement was observed between COG and MCNP results for ENDF/B-VIII.0; however, some of the uranium benchmark cases for JEFF-3.3 showed significant differences in the results, which may be attributed to discrepancies in the probability tables for the unresolved resonance region due to differences in processing the JEFF-3.3 data by BNL and LLNL as implemented in COG. A major inter-comparison project between COG, MCNP, MORET, and SCALE for ENDF/B-VIII.0 and JEFF-3.3 is also in progress. We anticipate that LLNL participation in this project will result in development of significantly more COG benchmark cases as our goal is to overlap the VALID, WHISPER, and IRSN compendia of criticality benchmarks to the extent possible, which will be beneficial to international COG, MCNP, MORET, and SCALE user communities.

Poster Session

TUESDAY, SEPTEMBER 17

16h10-17h50

Posters *Area*

N°	TITLE	AUTHORS
	Track 1. Codes and Other Calculation Methods	
1	Variations of the Effective Neutron Multiplication Factor Due to the Modelling of Granules and Boundaries in Generic Transport Packages Containing Volumes of Small Fissile Particles in KENO-VI/SCALE 6.2.1	**Dirk Schulze Grachtrup,** Benjamin Ruprecht, Frederik Kesting
2	Opera – A New Radiation Shielding Platform for Radiation Protection Studies and Criticality Accident Dose Assessment	**Arnaud Entringer,** Francis Kloss, Michael Laget, Fadhel Malouch, Hocine Oulebsir, Laurence Pangault, Daniele Sciannandrone, Thierry Visonneau
3	Results of Tripoli-4® Version 11 Code for Fast Spectrum Criticality Benchmarks	**François-Xavier Giffard,** Anne Mijonnet, Michaël Prigniau
4	Validation of MCNP6.1 and MCNP6.2 Using ENDF/B-VII.1 Nuclear Data for Criticality Safety Application to Plutonium and Highly Enriched Uranium Systems	**Shauntay. E. Coleman,** William J. Zywiec
	Track 2. Nuclear Data	
5	Benchmark Monte Carlo Calculations with ENDF/B-VIII and JEFF3.3 Libraries for LWR Criticality Safety Assessments	**Marco Pecchia,** Alexander Vasiliev, Hakim Ferrouki, Gregory Perret
6	Feedback on JEFF-3.3 and ENDF/B-VIII.0 Nuclear Data Using a Suite of Benchmarks from the MORET 5 Experimental Validation Database	**Nicolas Leclaire,** Luiz Leal, Frédéric Fernex
7	Validation of KENO V.A and KENO-VI in SCALE 6.3 Beta 3 Using ENDF/B-VII.1 and ENDF/B-VIII Libraries	**William J. Marshall,** Ellen M. Saylor, Andrew M. Holcomb, Dorothea Wiarda, Travis G. Greene
8	Analysis of D_2O Benchmark Criticality Experiments	**Travis M. Greene,** William J. Marshall, Guillermo I. Maldonado
	Track 3. Uncertainty and Sensitivity Analysis	
9	Analysis of Sufficiency of Benchmark Experiments during Validation of Nuclear Safety Calculations Programs	**Vladimir V. Tebin,** Dik T. Ivanov
10	Investigating Region-wise Sensitivities for Nuclear Criticality Safety Validation	**Bobbi Merryman,** Forrest Brown, Jennifer Alwin, Christopher Perfetti
11	Initial Application of TSUNAMI for Validation of Advanced Fuel Systems	**William J. Marshall,** Justin B. Clarity, Jinan Yang, Ugur Mertyurek, Matthew A. Jessee, Bradley T. Rearden

N°	TITLE	AUTHORS
Track 4. Measurements, Experiments and Benchmarks		
12	Growth of the International Criticality Safety and Reactor Physics Benchmark Experiment Evaluation Projects since ICNC 2015	**John D. Bess,** J. Blair Briggs, Tatiana Ivanova, Jim Gulliford, Ian Hill, Margaret Marshall, Lori Scott
13	Benchmark Evaluation of Water-Moderated Hexagonal Lattices 21% Enriched Uo2 Fuel Rods at "Rose" Critical Facility	**Svyatoslav Sikorin,** Andrei Kuzmin, Siarhei Mandzik, Tatsiana Hryharovich, Yuliya Razmyslovich
14	Critical Experiments on Zirconium Hydride-Moderated Hexagonal Lattices 45% Enriched Uo2 Fuel Assemblies at Crystal Critical Facility	**Svyatoslav Sikorin,** Siarhei Mandzik
15	Experimental and Calculated Data on Criticality of Hexagonal Lattices of 36 % Enriched Uranium Fuel Rods with and without the Boron Absorber Rods in Water	**Svyatoslav Sikorin,** Siarhei Mandzik, Siarhei Polazau, Andrei Kuzmin, Tatsiana Hryharovich, Yuliya Razmyslovich
16	CURIE Experiment: An Experiment to Validate and Test Updated URR Information	**Theresa Cutler,** Jesson Hutchinson, William Myers, Rian Bahran
17	Caliban Reactor Criticality Benchmark: Calculations and Interpretation of Simulations with Different Versions of TRIPOLI-4® and Different Nuclear Data Libraries	**Pierre Casoli,** Jean-Sébastien Borrod, Michaël Prigniau
Track 5. Standards, Assessment Methodology, Regulations		
18	Mixing Rule for Uranium and Plutonium Isotopes	**Georges Kyriazidis,** David Noyelles, Michaël Prigniau
Track 6. Operational Practices and Safety Cases		
19	Enhancement of Neutron Reflector Classification	**Aurélien Dorval,** David Noyelles, Marc Triballier, Michaël Prigniau
20	Criticality Safety Concept for Organic Additives Introduction in Granulation Process	**Nadine Comte,** Béatrice Thievenaz, Jean-François Paput
21	The Effect of Particle Size on the Reactivity of Powdered Fuels	**Albrecht Kyrieleis,** Ahmed Aslam, Andrew Thallon
22	Material Handling Store Concept Design	**J. Bell**, S. Plummer
Track 7. Storage and Transport issues		
23	A Parametric Study of the Effect of Reactor Operating Conditions on Gadolinium Peak Reactivity Determination for BWR UO2 Used Fuel Transport and Storage	**Marcel Tardy,** S. Kitsos, D. Lin, L. Milet, P. Puppetti, G. Grassi, V. Roland
24	Defining Safe Fissile Mass Limits for Transport Packages Carrying Intermediate Level Waste	**Daniel Fisher**
25	Evaluation of Criticality Safety Measures for Fuel Storage of Critical Assemblies in STACY	**Jun-Ichi Ishii,** Kazuhiko Izawa, Takuya Okubo, Kazuhiko Ogawa
Track 9. Criticality Accidents and Incidents		
26	Stochastic Behaviour of a Criticality Excursion with Low Source	**Philippe Humbert**
27	Review of Criticality Accident Alarm System Requirements in Geological Disposal	**Dr Liam Payne,** Neil Harris
Track 10. Professional Development Issues and Training		
28	Little Criticality: A Helpful Tool for a Criticality Safety Engineer	**Aurélien Poisson,** Steve Duquenne, Ilyes Bouaoud, Alexandre Coulaud, Julie Jaunet
29	Neutronics Livened Up by Computer	**Paul Reuss**
30	Criticality Safety Training Approach in a Fuel Assembly Manufacturing Site: Testimony and Considerations	**Jean-François Paput**

VARIATIONS OF THE EFFECTIVE NEUTRON MULTIPLICATION FACTOR DUE TO THE MODELLING OF GRANULES AND BOUNDARIES IN GENERIC TRANSPORT PACKAGES CONTAINING VOLUMES OF SMALL FISSILE PARTICLES IN KENO-VI/SCALE 6.2.1

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For packages designed to contain fissile powder or which may contain small fissile particles the criticality safety assessment needs to address the effects of particle sizes, spacing, moderation medium and others. Often this is done using only one specific modelling approach which is well established within the company or institution in charge while other approaches are not addressed. Here, we present a detailed study of the impact of different modelling approaches for small water-moderated particles in infinite and finite systems for different fuel compositions and geometries. Fuel compositions and finite geometries were chosen to resemble typical fissile media and transport packages actually used in Europe. We compare homogenized and different explicit models, with or without partially cut granules at the edges, as well as continuous energy and multigroup Monte Carlo calculations of identical configurations. At system sizes chosen to result in k_{eff} » 0.95, where small numerical differences might be crucial to obtain official approval for package designs, we observe distinct differences up to more than 1% in k_{eff} depending on subtleties of the calculational scheme, fissile medium, granule cutting and reflecting material of the generic package. For package designs close to the administrative limit awareness of such effects should be present.

OPERA – A NEW RADIATION SHIELDING PLATFORM FOR RADIATION PROTECTION STUDIES AND CRITICALITY ACCIDENT DOSE ASSESSMENT

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OPERA, acronym standing for « Outil Plateforme pour les Études de RAdioprotection », is purposed to be an integrated radiation protection computation software platform allowing users to model a study once and then proceed calculations with different methods. The intrinsic link between criticality accident and shielding, as well as the integration of the massively used in criticality calculation code TRIPOLI-4®, shows that OPERA can be used for some aspects of criticality safety studies, even if its initial purpose is shielding calculations.

The major improvement allowed by the initial technical specifications of the OPERA platform is to define only once every necessary parameters to fully describe a study. The user will then be able to choose which calculation route he wants to use to solve his problem. Three routes will be implemented in the platform, each being identified by the aliases "simplified", "industrial" and "reference". The simplified route relies on fast approximate codes to obtain first order estimates in order to support pre-design studies. The industrial route involves code dealing with more realistic physics model such as 2D/3D deterministic codes or multigroup Monte Carlo. The third one, called "reference" is the route providing the use of the French continuous energy Monte Carlo reference code TRIPOLI-4®. These details apply to transport problems, but depletion/activation codes are also scheduled to be integrated in each route (Cf. Figure 1).

These three routes allow obtaining results from a first order estimate, aimed to be conservative, to a more accurate result without each time entirely re-describing the study, and in the process, ensuring consistency. This is permitted by using another route to obtain additional physical parameters when needed. All relevant data are transmitted seamlessly to the different codes.

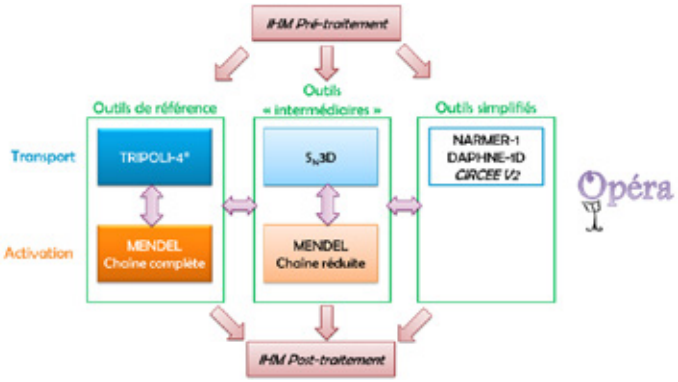
These routes are the basis of the calculation schemes that will be available in the platform. Those schemes will be representative of schemes usually needed to perform shielding calculations (activation, propagation of a secondary source ...).

Within these different schemes, one, dedicated to criticality accident dose estimations, is under development by the French Alternative Energies and Atomic Energy Commission. It is intended to allow fast assessment of operational dose over complex geometries and dosimetric consequences of a criticality accident.

Two examples will drive the further criticality safety part implementation in OPERA. The first one is a usual dose map problem fully modeled and compared to a reference calculation. The second use case is an ICSBEP problem of shielding, modeled using the simplified route and by the reference route using the automated deterministic importance map calculation to speed up the reference Monte Carlo calculation performed by the TRIPOLI-4® code.

The first operational version of OPERA is scheduled for the end of 2020.

Figure 1. Different routes within the platform



RESULTS OF TRIPOLI-4® VERSION 11 CODE FOR FAST SPECTRUM CRITICALITY BENCHMARKS

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This paper presents the results of the latest version of the TRIPOLI-4® Monte Carlo code from its criticality validation database. This criticality database is made of 1350 selected ICSBEP benchmarks with various enrichments, fissile materials, spectra and non-fissile media (reflectors, absorbing media). The ICSBEP TRIPOLI-4® benchmarks computational models are prepared under the responsibility of the CEA. This paper is focused on fast spectrum benchmarks for uranium and plutonium fissile media. Comparisons are made with the experimental (benchmark model) k_{eff} given in the ICSBEP documents. For HEU-MET-FAST category, the mean discrepancy is equal to -0.00178 for a standard deviation equal to 0.00408. Good agreements are also obtained for PU-MET-FAST category with a mean discrepancy equal to -0.00006 for a standard deviation equal to 0.00462.

VALIDATION OF MCNP6.1 AND MCNP6.2 USING ENDF/B-VII.1 NUCLEAR DATA FOR CRITICALITY SAFETY APPLICATION TO PLUTONIUM AND HIGHLY ENRICHED URANIUM SYSTEMS

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Monte Carlo radiation transport codes have become the preferred tool for engineers in writing criticality safety evaluations, but in order to use these codes, engineers must understand and be confident in the results produced. This is one of the reasons verification and validation of software codes is required per ANSI/ANS-8.1 and ANSI/ANS-8.24. At Lawrence Livermore National Laboratory (LLNL), validation has been completed for MCNP6.1 for plutonium and highly enriched uranium systems by using a validation suite of 1,196 experimental benchmark cases (488 plutonium and 708 highly enriched uranium) that were published in the ICSBEP Handbook.. The keff and standard deviation results from the validation of MCNP6.1 were then compared against preliminary validation results from MCNP6.2, using cross sections based on the ENDF/B-VII.1 nuclear data library and the same validation suite of 1,196 cases. Overall, the results showed an agreement of 92.6%, indicating only 88 cases between plutonium and highly enriched uranium had differences between the two versions of the code. Although MCNP6.2 fixed over 300 "bugs" in the code and has numerous updates to the coding and nuclear data libraries, less than one-twentieth of those fixes affected criticality safety calculations. By researching the changes to MCNP6.2 in detail, it was concluded that only three major changes affected LLNL:

- (1) updates and corrections to nuclear data libraries,
- (2) coincident surface treatment correction,
- and (3) a new Fortran complier. However, each of the 88 cases was within one standard deviation.

Since the MCNP random number generator is not strictly random and uses the same sequence number for an identical input deck to reproduce a sequence, it is expected with so few changes to MCNP6.2 for nuclear criticality safety that the keff and standard deviation should be similar if not identical. The high agreement between the MCNP6.1 and MCNP6.2 results, even with code updates and "bug" fixes, shows good comparison in code performance. With less than a 10% change in the calculated values, the upper subcritical limit between MCNP6.1 and MCNP6.2 for both plutonium and highly enriched systems are identical to four decimal places. The information presented in this paper will be used to complete the final validation for MCNP6.2 for plutonium and highly enriched uranium applications on LLNL workstations.

BENCHMARK MONTE CARLO CALCULATIONS WITH ENDF/B-VIII.0 AND JEFF-3.3 LIBRARIES FOR LWR CRITICALITY SAFETY ASSESSMENTS

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A validation of the PSI Criticality Safety Evaluation methodology using the most recent evaluated nuclear data library ENDF/B-VIII.0 and JEFF-3.3 is presented in this paper. The basis for the methodology is a well-defined benchmark suite of 149 low-enriched thermal compound uranium critical experiments contained in the International Handbook of Evaluated Criticality Safety Benchmark Experiments. The validation of code/libraries is focused on the evaluation of the calculated-to-experimental results lower tolerance bound along with trend analyses for both design parameters as well as nuclear spectrum related parameters. The overall performance of the new libraries for these selected thermal systems is evaluated with the neutron transport codes MCNP® and SERPENT, the latter used only with the JEFF-3.3 library for assessing also the code related uncertainty. Trends as function of design-based parameters and/or spectral parameters are analysed, focusing on certain groups of experiments in which the specific library changes have the largest impact on the evaluated differences. Finally, a numerical study about the effect of the Monte Carlo precision to the κ_{eff} distribution is also presented, highlighting that the shape of the histogram drawn from the normalized κ_{eff} values, collected over the 149 benchmark cases, depends on the Monte Carlo precision level. This behaviour will be investigated to find a sufficient level of Monte Carlo precision for which the hypothesis of Normality cannot be refused and the upper subcritical κ_{eff} limit can be derived using parametric statistics.

FEEDBACK ON JEFF-3.3 AND ENDF/B-VIII.0 NUCLEAR DATA USING A SUITE OF BENCHMARKS FROM THE MORET 5 EXPERIMENTAL VALIDATION DATABASE

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It has been more than ten years now since IRSN began developing the MORET 5 Monte Carlo code. This code is used in several applications, in particular:

- Within the CRISTAL package, coupled with the deterministic APOLLO2 cell code that makes flux calculation and generates homogenized, self-shielded cross sections for use in the MORET 5 code that performs the 3D calculation,
- In a continuous energy version, using nuclear data at the ACE format processed by the GAIA tool based on NJOY 2016.

The last version of the code, named MORET 5.D.1, has been validated for internal use with about 1300 experiments mainly coming from the ICSBEP Handbook. This database covers almost all operations of the fuel cycle encountered in criticality-safety studies. It therefore constitutes an opportunity to test nuclear data libraries. In particular, it has been used for testing JEFF-3.2, all intermediate versions of JEFF-3.3, as well as the ENDF/B-VIII.0 evaluation. The results of the testing were reported by IRSN at the JEFF meetings. For that purpose, the associated nuclear data libraries were generated using the GAIA 1 tool developed by IRSN, which encapsulates the NJOY 2016 software and brings additional tests to check consistency of produced nuclear data. It converts ENDF nuclear data files in ACE formatted files that can be easily read by the MORET 5.D.1 code through an xml file indicating the address of files.

The objective of the paper is to show the feedback on nuclear data that such a database can bring through a selection of dedicated cases, where significant discrepancies between evaluations are observed. For that reason, sensitivity calculations to the main reactions are performed in order to identify the energy zones where improvements of cross sections could have an impact on κ_{eff} results. Sensitivity/uncertainty tools such as TSUNAMI/TSURFER or MACSENS are also used in keeping with the GLLSM methodology for assimilation of nuclear data in order to identify potential adjustment of cross sections. Biases due to nuclear data might then be determined for industrial applications whose k_{eff} is also sensitive to the same elements as the experiments of the database.

VALIDATION OF KENO V.A AND KENO-VI IN SCALE 6.3 BETA 3 USING ENDF/B-VII.1 AND ENDF/B-VIII LIBRARIES [1]

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In early 2019, internal beta releases of SCALE 6.3 were identified for internal users, as well as a limited number of external users, to apply when testing key features. Recently developed AMPX-formatted libraries based on ENDF/B-VIII have also been generated for use in internal testing. In this effort, a criticality safety validation was performed using nuclear data libraries based on the ENDF/B-VII.1 and ENDF/B-VIII releases to quantify the changes caused by code development and to identify differences caused by the new nuclear data. As with past validations, the primary suite of experiments used is the Verified, Archived Library of Inputs and Data (VALID) maintained in the Reactor and Nuclear Systems Division at Oak Ridge National Laboratory. The VALID suite contains a total of 618 individual configurations, including systems with fast, thermal, mixed, and intermediate neutron spectra. Fissile species include a range of uranium enrichments, plutonium, mixtures of uranium and plutonium, and ^{233}U. Metal,

solution, and compound, mostly in pin array fissile forms, are included in the suite. The results presented here include the 252-group general purpose multigroup library based on ENDF/B-VII.1 and the continuous-energy libraries based on ENDF/B-VII.1 and ENDF/B-VIII.

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ANALYSIS OF D_2O BENCHMARK CRITICALITY EXPERIMENTS

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An analysis of systems containing deuterium was conducted using ICSBEP Handbook cases to address two areas of concern: first, prior reports of discrepancies for deuterium-moderated experiments with ENDF/B-VII.0 cross sections, and second, difficulties encountered in SCALE with intermediate spectrum systems. Consequently, 89 cases from 16 evaluations utilizing deuterium as either a moderator or reflector were modeled using KENO V.a/VI, MCNP 6.1.1b, and SERPENT 2, with continuous energy (CE) and multigroup (MG) libraries (CE ENDF/B-VII.1, CE ENDF/B-VII, 252-group ENDF/B-VII.1). The models encompass differences in neutron energy (thermal, intermediate/mixed, fast), enrichment of $^{235/233}U$ (low, intermediate, high), and fissile material (solution, composite, metal). All cases were evaluated by C/E ratio and energy of average lethargy of fission (EALF).

The C/E values across all cases indicate a bias below 3.0%Δk with an average bias of 0.77%Δk. However, if the mixed/intermediate neutron energy spectrum cases are removed, the bias decreases to 1.83%Δk with an average of 0.57%Δk. No single code/library combination appears to outperform another as each differing combination arrives at similar results. While only ENDF/B-VII.1 was used to compare MG and CE libraries, an examination of the results indicates a wide range in the discrepancies between the two cross section libraries, with differences ranging from a maximum of 922 pcm to a minimum of 4.6 pcm, and an average of 191 pcm.

Among all codes and libraries for each experiment within a case, the two largest differences occur in U233-COMP-THERM-004-001 (UCT), with a 961 pcm difference between MCNP and KENO, and in HEU-COMP-THERM-018-001 (HCT), with a 922 pcm difference between MG-KENO and CE-KENO. If UCT-004 and HCT-018 are excluded, the remaining 62 experiments show differences of less than 500 pcm with an average difference of 195 pcm.

Overall, the codes and libraries used for modeling these experiments tend to slightly overestimate LEU, metal, and thermal systems, while underestimating HEU, solution, compound, fast, and intermediate systems. The models here provide consistent results with variable uncertainties across a wide range of systems that utilize deuterium either as a moderator or reflector. The results form the basis for further exploration into deuterium systems. Other experiments could provide a more rigorous and complete picture of deuterium nuclear data performance.

Track 3

ANALYSIS OF SUFFICIENCY OF BENCHMARK EXPERIMENTS DURING VALIDATION OF NUCLEAR SAFETY CALCULATIONS PROGRAMS

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In the current report determination of standard deviation of calculation bias (difference between experimental and calculated k_{eff}) and normality of distribution of calculation biases and statistical analysis for the following sets of benchmarks from ICSBEP Handbook is presented:sets of benchmarks classified by volumes of ICSBEP Handbook; sets of benchmarks classified by hardness of the neutron spectrum and the types of experiments;sets of benchmarks classified by years of publication in ICSBEP Handbook; The results of statistical processing show almost complete normality of the distribution of calculation biases of the benchmarks, which were added to ICSBEP Handbook before 2009, excluding four types of benchmarks (assemblies with an intermediate spectrum, assemblies with a thermal spectrum and an organic moderator, homogenized ZPR-type models, subcritical assemblies). For benchmark introduced after 2009 (excluding same set of data) following changes were observed: with a decrease in the average difference between experimental and calculated k_{eff}, there is a noticeable deviation of the shape of the distribution from the normal distribution. These changes can be considered as artificial and caused by selection of "good" experiments from totality of experiments made by its authors before being sent to the ICSBEP. It should be noted that not only selection of experiments by author can lead to the appearance of distributions thinner than the standardized normal (respectively to a higher maximum), but also the manual preparation of the initial data. Based on results of statistical processing following conclusions can be made: benchmarks which were added to ICSBEP Handbook before 2009 form sufficient set and thus can be used for validation of programs via statistical methods; benchmarks which were added to ICSBEP Handbook after 2009 as well as calculated resuls which were published in the same period should be used in statistical validation carefully after checking normality of distribution of standartisezed differences between calculated and experimental data

INVESTIGATING REGION-WISE SENSITIVITIES FOR NUCLEAR CRITICALITY SAFETY VALIDATION

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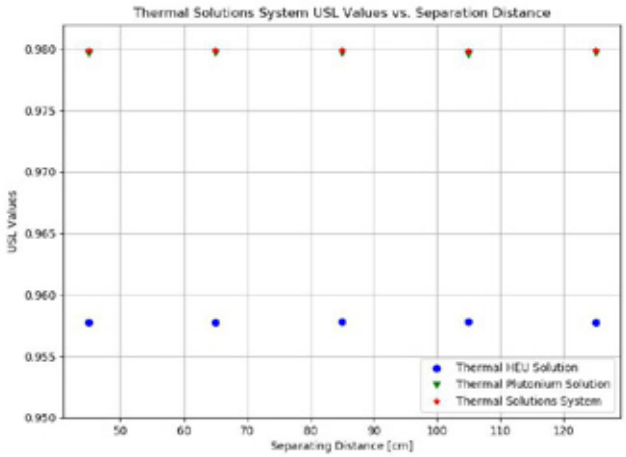
Criticality safety analysts estimate Upper Subcriticality Limits (USLs) for subcritical systems to account for biases and errors in modeling and simulation tools and to ensure that conditions calculated to be subcritical will actually be subcritical This study investigates the application of the MCNP6-Whisper [1,2] calculational methods to estimate USLs for loosely-coupled systems comprised of two units and makes use of a new MCNP6 capability that allows for the calculation of multiple region-dependent sensitivity profiles in a single calculation [3]. MCNP6 can compute the region-wise sensitivity profiles for each unit, as influenced by the leakage of the other unit, and the sensitivity profile of the overall loosely-coupled system. This investigation deliberately focused on USL estimates for small, simple systems to highlight physical and computational issues associated with these types of systems. Three basic highly enriched uranium (HEU) and plutonium components are used in the loosely-coupled models in this study: bare fast metal sphere, water-reflected fast metal sphere, and thermal solution. These three different units are paired in various combinations, and the MCNP6-Whisper calculations are performed using both the new region-wise sensitivity profile capability and the conventional overall sensitivity profile capability. This study evaluates the Whisper-selected benchmark rankings and calculated baseline USL values for each calculated sensitivity profile at each separation distance for each application model.

Figure. 1 Thermal Solution System's baseline USL values as a function of unit separation distance.



The thermal solution model baseline USL values determined by Whisper are of great interest, as seen in Figure 1. The USL values for the overall application model closely match the USL values for the thermal plutonium solution assembly, leading to nonconservative USL values for the overall application model.

Figure 2. Mixed Plutonium System's baseline USL values versus unit separation distance.

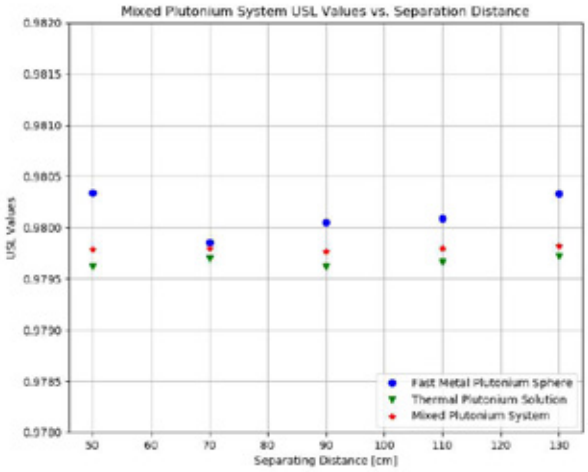


Table I. Benchmark rankings for Fast Metal Plutonium Unit in Mixed Plutonium Unit

Fast Metal Plutonium Assembly	Separation Distance [cm]				
Benchmark Ranking	50	70	90	110	130
1	pmf-044-004.i	pmf-042-002.i	pmf-044-004.i	pmf-044-004.i	pmf-044-004.i
2	pmf-044-005.i	pmf-042-003.i	pmf-044-005.i	pmf-044-005.i	pmf-044-005.i
3	pmf-024-001.i	pmf-044-005.i	pmf-042-002.i	pmf-042-002.i	pmf-031-001.i
4	pmf-036-001.i	pmf-027-001.i	pmf-042-001.i	pmf-031-001.i	pmf-024-001.i
5	pmf-031-001.i	pmf-044-004.i	pmf-031-001.i	pmf-042-001.i	pmf-036-001.i
6	pmf-044-003.i	pmf-042-001.i	pmf-042-003.i	pmf-042-003.i	pmf-042-001.i
7	pmf-042-002.i	pmf-042-004.i	pmf-027-001.i	pmf-027-001.i	pmf-044-003.i
8	pmf-042-001.i	pmf-011-001.i	pmf-011-001.i	pmf-042-004.i	pmf-042-002.i
9	pmf-027-001.i	pmf-042-005.i	pmf-042-004.i	pmf-036-001.i	pmf-027-001.i
10	pmf-042-004.i	pmf-031-001.i	pmf-036-001.i	pmf-011-001.i	pmf-011-001.i

For the mixed plutonium system, there is noticeable variation in the USL values for the fast-reflected metal plutonium unit with respect to separation distance, seen in Figure 2 and Table 1. This suggests that the energy spectra of the neutron leakage from each unit, as a function of separation distance between units, may be of influence in systems with units of dissimilar neutron energy spectra and of negligible influence in systems where units are of similar neutron energy spectra.

This numerical experiment highlights that nonconservative USL estimates may be calculated for loosely-coupled multi-region critical models when using only the overall sensitivity profile of systems. This work suggests that to ensure conservative USL estimates for loosely-coupled systems, both the overall system's sensitivity profiles and region-wise sensitivity profiles should be calculated and utilized by Whisper 1.1.

[1] F.B. Brown, M.E. Rising, J.L. Alwin, "User manual for whisper-1.1," LA-UR-17-20567 (2017)

[2] F.B. Brown, M.E. Rising, J.L. Alwin, "Release Notes for whisper-1.1,", LA-UR-17-23504 (2017)

[3] B. Merryman, F.B. Brown, J.L. Alwin, "Investigating Region-wise Sensitivities for Nuclear Criticality Safety Validation, LA-UR-18-31601 (2017)

INITIAL APPLICATION OF TSUNAMI FOR VALIDATION OF ADVANCED FUEL SYSTEMS [1]

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A number of advanced reactor concepts and accident-tolerant fuels (ATFs) are under development or in the initial stages of deployment and testing in the nuclear power industry. Many of these concepts incorporate chemical or material forms that have not traditionally been used in the nuclear industry and may present challenges for nuclear criticality safety validation. Sensitivity/uncertainty methods are ideally suited for assessing the applicability of existing critical experiments to these potentially challenging applications. This paper presents two case studies recently completed at Oak Ridge National Laboratory (ORNL) which investigated the use of the TSUNAMI tools within the SCALE code system to perform such assessments.

Many advanced reactor concepts include uranium enrichments above 5 wt% ^{235}U but below the 20 wt% limit of low-enriched uranium (LEU). This material, referred to as high assay LEU (HA-LEU), poses difficulties across the entire infrastructure of nuclear power, including enrichment, transportation, fabrication, and storage. A potentially representative transportation package was selected to perform a criticality safety validation assessment for UF_6 containing HA-LEU. Sensitivity data were generated using the TSUNAMI-3D sequence, and an assessment of the applicability of a range of critical experiments was performed using the TSUNAMI-IP tool. The critical experiments were drawn from the set of available sensitivity data files distributed with the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Handbook. The results indicate that a sufficient number of available experiments may be available to perform validations for these systems.

Many different cladding materials and fuel dopants are included in ATFs to mitigate the consequences of a loss of coolant accident. Some of these systems have already been fabricated and introduced as lead test assemblies in commercial reactors, while others are still being developed. Several concepts were considered for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) in comparison to the current Zircaloy and UO_2 fuel system. In most of these cases, the ATF materials did not significantly impact the number of applicable experiments for validation. Some fuel and cladding materials were identified, however, which may reduce the number of available experiments and which will increase the nuclear data–induced uncertainty in neutron multiplication.

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Track 4

GROWTH OF THE INTERNATIONAL CRITICALITY SAFETY AND REACTOR PHYSICS BENCHMARK EXPERIMENT EVALUATION PROJECTS SINCE ICNC 2015

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Since ICNC 2015 there has been continued success with the OECD NEA sanctioned international benchmark projects: the International Criticality Safety Benchmark Evaluation Project (ICSBEP) and the International Reactor Physics Experiment Evaluation Project (IRPhEP). The most recent contributions are summarized herein; more significant details regarding each benchmark evaluation are contained within either the International Handbook of Evaluated Criticality Safety Benchmark Experiments and the International Handbook of Evaluated Reactor Physics Benchmark Experiments. The content within these two handbooks continues to expand to provide high-quality integral benchmark data of use to the criticality safety, nuclear data, and reactor physics communities.

BENCHMARK EVALUATION OF WATER-MODERATED HEXAGONAL LATTICES 21% ENRICHED UO2 FUEL RODS AT "ROSE" CRITICAL FACILITY

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Criticality experiments for four hexagonal lattices of fuel rods in water were performed using "Rose" critical facility at the Joint Institute for Power and Nuclear Research – Sosny of the National Academy of Science of Belarus. The fuel rods pitches of the different arrangements are 10.5, 18.19, 21.0 and 27.78 mm with a triangular grid. The fuel composition – UO_2 with 21% uranium-235 enrichment. The moderator and reflector - H_2O. The fuel rods are completely flooded with water and the number of rods was adjusted to reach criticality. The results of experiments received on critical assemblies have been analyzed by means of detailed calculation models development and performing numerical simulations of the experiments using the MCU-PD Monte Carlo computer code for this purpose. In addition, the sensitivity and uncertainty analysis of the experiments has been performed. In this paper, both the experimental and numerical results for four critical configurations of "Rose" facility as well as the associated uncertainty analysis are presented. Uncertainty treatment demonstrates that the experiments are good candidates for criticality benchmarks with thermal neutron spectrum and uranium dioxide fuel having intermediate enrichment.

CRITICAL EXPERIMENTS ON ZIRCONIUM HYDRIDE-MODERATED HEXAGONAL LATTICES 45% ENRICHED UO_2 FUEL ASSEMBLIES AT CRYSTAL CRITICAL FACILITY

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Experiments on criticality of uranium-zirconium hydride systems were performed on the critical facilities "Edelveis", GFS, "Crystal" and "Giacint" of the Joint Institute for Power and Nuclear Research – Sosny of the National Academy of Sciences of Belarus. Critical facilities were designed to simulate different reactor core configurations for development of new reactors of different functions and benchmark experiments. The experiments were performed with 21%, 36% and 45% enriched UO_2 fuels. A lot of experiments were performed for validation and verification of the computer codes used for neutron calculations of uranium-zirconium hydride multiplication systems.

The critical facility "Crystal" was used investigate the characteristics of numerous critical assemblies configurations with various modifications of the cores, representing non-uniform multiple zones heterogeneous uranium-zirconium hydride lattices comprising hexagonal fuel assemblies with cylindrical fuel rods, boron absorbing plates, boron and europium absorbing rods, zirconium hydride and steel side and end reflectors. The critical assemblies represented the cores collected from three types of fuel assemblies with different structure, surrounded by assemblies and units of a side reflector. The core included channels for the regulating, compensating and emergency protection rods. The moderator – $ZrH_{1.9}$. The fuel composition – UO_2-Ni-Cr matrix with 45% uranium-235 enrichment. The absorber in plates – B with 85% boron-10 enrichment. The absorber in rods – B_4C and Eu_2O_3. We measured the critical configurations, the reactivity margin, the efficiency of the absorber rods and the dependence of the efficiency of the absorber rods on their depth of insertion in a core. Different methods for measuring the reactivity, the reactivity margin and the efficiency of the absorber rods was used. The paper provides the description of the structure and the composition of the investigated critical assembly configurations and the associated experimental results.

EXPERIMENTAL AND CALCULATED DATA ON CRITICALITY OF HEXAGONAL LATTICES OF 36 % ENRICHED URANIUM FUEL RODS WITH AND WITHOUT THE BORON ABSORBER RODS IN WATER

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Criticality experiments for hexagonal lattices of fuel with and without absorbing rods in water were performed using "Giacint" critical facility of the Joint Institute for Power and Nuclear Research – Sosny of the National Academy of Science of Belarus. The fuel rods pitch is 25 mm. The absorber rods pitch is 50 mm. The fuel composition – UO_2 with 36% uranium-235 enrichment. The absorber composition – natural B_4C. The active fuel and absorber length is 500 mm. Clad material of rods is stainless steel. The moderator and reflector – H_2O. The criticality conditions obtained by adjusting the water moderator height in the studied lattices. The results of experiments on critical assemblies have been analyzed by creating detailed calculation models and performing simulations for the experiments. The analyses used the MCNP-4C and MCU-PD computer programs.

CURIE EXPERIMENT: AN EXPERIMENT TO VALIDATE AND TEST UPDATED URR INFORMATION

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The Unresolved Resonance Region in Uranium has been extensively studied over the past decade. This is largely attributed to the significant increases in computing power. It is now feasible to calculate and use less approximations in this region. The studies over the past decade have primarily been in differential experiments and computer codes and simulations.

With the increased knowledge of the URR, especially in uranium, there is a need for integral experiments that are sensitive to this energy region. This is a step beyond the extensive research on the intermediate energy region, which started at Los Alamos National Laboratory in the 1990s. Given LANL's history of conducting experiments and knowledge of the behavior of integral benchmarks, LANL will perform integral experiments that are sensitive to the URR in with highly enriched uranium. These experiments have been designed over the past two years and will be executed in the coming year. The work will help validate new evaluations and updates to nuclear data libraries based on recent differential measurements conducted at Rensselaer Polytechnic Institute (RPI) and LANL. It is also based on recent code developments that have been included in OpenMC. The integral experiments have been designed with the utility of expertise from people who have extensively analyzed the original intermediate energy ZEUS benchmark series conducted at LANL

The integral experiments will be conducted at the National Criticality Experiments Research Center (NCERC) at the Nevada National Security Site in Nevada, USA. Experiments will be performed on the Comet vertical lift assembly machine. The setup is generally as follows: thick copper reflector and uranium cylindrical plates with teflon plates interleaved. The copper is based on analysis done in the 1990s which show that a moderate Z reflector will scatter some neutrons back to the core without significant energy loss. The HEU plates are known as the Jemima plates- they are large thin cylinders of bare material. The purpose of the teflon is to optimize the number of fissions in the HEU in the U-235 URR. Upon completion of these experiments, the data will be analyzed and the analysis will be submitted for inclusion in the International Criticality Safety Benchmark Evaluation Project Handbook.

CALIBAN REACTOR CRITICALITY BENCHMARK: CALCULATIONS AND INTERPRETATION OF SIMULATIONS WITH DIFFERENT VERSIONS OF TRIPOLI-4® AND DIFFERENT NUCLEAR DATA LIBRARIES

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Criticality benchmarks are reference experiments performed in particular to test and compare different neutron transport codes and different evaluated nuclear databases. Caliban reactor was a cylindrical critical assembly, with a highly enriched uranium metallic core. It was operated at CEA Valduc center until the beginning of the 2010's. Thanks to its simple geometry and to its fuel composition, Caliban was a device adapted for reference experiments. A criticality benchmark using Caliban reactor was then performed and published in 2007, in the International Handbook of Evaluated Criticality Safety Benchmark Experiments [1].

In this Handbook, the chosen experiment was modeled with codes and libraries available at that time: TRIPOLI-4.4.1® code [2] [3] with ENDF/B-6 R4 library [4], MCNPX-2.6 code [5] with ENDF/B-6 R0 library [4] and MCNP-5 code [6] with ENDF/B-6 R6 library [4].

Transport codes and evaluated data have been improved since the benchmark publication. In this paper, the model published in the benchmark is run with new tools, especially with the last available version of TRIPOLI-4® code and the last available version of ENDF library, but also with a recent version of JEFF data library.

Simulation results are compared with published Caliban benchmark calculations and differences are discussed.

This work is also an opportunity to check the consistency between the experiment description and the benchmark model, both published in the Handbook: geometry dimensions, mass and density definitions of the different model elements are particularly studied. Consistency between the benchmark model and the modeling using the simulation codes are studied too: the definition of the volumes are especially checked.

[1] N. Authier, B. Mechitoua, P. Grivot, P. Humbert and N. Ellis, "HEU-MET-FAST-080, Bare, Highly Enriched Uranium Fast Burst Reactor Caliban (September 30, 2007), NEA/NSC/DOC/(95)03/II, Volume II", International Handbook of Evaluated Criticality Safety Benchmark Experiments, NEA (December 2016).

[2] E. Brun, F. Damian, C.M. Diop, E. Dumonteil, F.X. Hugot, C. Jouanne, Y.K. Lee, F. Malvagi, A. Mazzolo, O. Petit, J.C. Trama, T. Visonneau, and A. Zoia, "TRIPOLI-4®, CEA, EDF and AREVA reference Monte Carlo code", Annals of Nuclear Energy, 82, pp.151–160 (2015).

[3] O. Petit, F.X. Hugot, Y.K. Lee, C. Jouanne and A. Mazzolo, "TRIPOLI-4 version 4 Manuel de l'utilisateur", Rapport CEA-R-6170 (2008).

[4] V. McLane et al, "ENDF-201,ENDF/B-VI Summary Documentation, Supplement I, ENDF/HE-VI Summary Documentation", NNDC, BNL (December 1996)

[5] J.S. Hendricks, et al., «MCNPX 2.6.0 Extensions», LA-UR-2216 (2008).

[6] X-5 Monte Carlo Team, "MCNP - A General N-Particle Transport Code, Version 5, Volume I: Overview and Theory", LA-UR-03-1987 (2003, updated 2005).

Track 5

MIXING RULE FOR URANIUM AND PLUTONIUM ISOTOPES

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Operations performed in nuclear research and development facilities using mass as a criticality control parameter, often consider uranium or plutonium isotopes exclusively as ^{239}Pu, which is a very conservative approach. However, the amount of the handled fissile material can be significantly enhanced using the following mixing rule between uranium fissile isotopes and ^{239}Pu isotope, calculated by the "usual" rule of fractions" (ROF) formula:

$$\frac{m\left(U\left(93.5\%\ wt.\ ^{235}U\right)\right)}{610}+\frac{m\left(^{239}Pu\right)}{350}\leq 1$$

The values 610 g and 350 g are the "safe" values obtained when multiplying the "old" critical values (keff = 1) of ^{235}U and ^{239}Pu by the usual "safety factor" 0.7.

This paper presents the demonstration of the validity of the aforementioned following mixing rule:

$$\frac{m\left(^{233}U\right)}{391}+\frac{m\left(U\left(93.5\%\ wt.\ ^{235}U\right)\right)}{596}+\frac{m\left(^{239}Pu\right)}{357}\leq 1$$

Calculations were performed using the APOLLO2-Sn route of the French criticality package CRISTAL V2. The values 391 g, 596 g and 357 g are the new "safe" values obtained through the new critical values calculated by CRISTAL V2.

Track 6

ENHANCEMENT OF NEUTRON REFLECTOR CLASSIFICATION

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This paper presents an enhanced classification for more than twenty reflector materials of variable thickness, placed around a moderated or unmoderated sphere of uranium (with low or high enrichment in ^{235}U) or plutonium fissile material. Eventually, a layer of water placed around the resulting sphere permits to evaluate the impact on this classification.

Calculations are performed using APOLLO2-Sn route of CRISTAL V2, using the bias reduction procedure that produces calculation results greatly improved when compared to results obtained with the TRIPOLI-4® calculation route.

CRITICALITY SAFETY CONCEPT FOR ORGANIC ADDITIVES INTRODUCTION IN GRANULATION PROCESS

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Current refurbishments of the Framatome LEU-UO_2 fuel fabrication facility in Romans France include an upgraded powder treatment process. These upgrades are planned to decrease the process cycle time leading to cost saving. One of them involves the powder densification and granulation processes.

After conversion from UF_6, UO_2 powder is pneumatically transferred into series of hoppers. From these hoppers the powder is introduced into a blender to be homogeneously mixed with pore-former and lubricant. Theses organic additives (and U_3O_8) are fed through other hoppers.

The blender may contain several hundred kg of fissile material and criticality is prevented by two control modes: mass and moderation.

Mass control mode is ensured by an online weighing device on the hoppers and the blender.

Moderation control mode is ensured by the hopper volume and by double-weighing of additives, guaranteeing the acceptable hydrogen content versus the fissile material batch mass.

In a first step when the additives are introduced, the blender is not running and the two media (fissile material and additives) may not be fully homogeneously mixed. This transient phase

disappears with the blender start up. During this transient phase, the acceptable hydrogen content has to be mixed with a limited fissile material weight leading to a maximum of reactivity and so called heterogeneous moderation.

This paper presents firstly the related methodology to determine the maximum permissible hydrogen mass as a function of the UO_2 powder mass and after that, the application of the permissible hydrogen mass obtained in the criticality-safety analysis.

Besides, the product obtained at the end of the granulation process must respect a maximum content in moisture equivalent. It leads to define maximum proportions of additives (hydrogen quantity per kg of powder) in different stages of the process. This last requirement finally leads to limit the quantity of hydrogen to a lower value and thus guarantees the criticality safety of the equipment.

The French criticality code package CRISTAL V1 was used to perform the calculations. This package consists of two calculation routes, using basic nuclear data based on the JEF2.2 evaluation:

- a standard route based on group-wise cross sections using APOLLO2 and MORET 4 codes,
- a reference route using the point-wise cross section and continue energy code TRIPOLI4®.

APOLLO2 is well suited to perform criticality standards calculations since it includes a sophisticated self-shielding approach, a Pij flux determination, and a 1D transport (Sn) process, and a great saving of calculation CPU time versus the reference scheme.

THE EFFECT OF PARTICLE SIZE ON THE REACTIVITY OF POWDERED FUELS

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Water moderated, low enriched uranium systems are known to have their highest neutron multiplication when the fissile and moderator materials are arranged heterogeneously; higher uranium enrichment or plutonium systems are often assumed to demonstrate their greatest neutron multiplication when arranged homogeneously. Recent work performed by the authors has established that for certain powdered fuels (PuO_2 and UO_2 with >10%wt ^{235}U) heterogeneous systems can result in higher neutron multiplication than equivalent homogeneous systems under certain conditions. A study using MONK has investigated the effect of particle size on neutron multiplication and its dependence on various parameters such as concentration, temperature, enrichment and nuclear data. The results of these studies are discussed in this paper.

MATERIAL HANDLING STORE CONCEPT DESIGN

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A vacant concrete civil structure is currently being assessed for use as a fissile Material Handling Store (MHS). The MHS will be expected to store Enriched Uranium in a configuration utilising Savy4000TM containers.

To support this activity, calculations were carried out to inform the development of Concept Design options for the MHS. Specifically, this was to determine the safe separation of storage containers for 1 or 2-high arrangements, whereby the MHS will remain safely subcritical during both normal operations as well as a range of fault scenarios.

These safe separations were then compared and used to calculate the maximum number of containers that could be stored within the available floor space of the MHS.

The fault conditions that were specifically assessed included over-batching, use of excessive packaging material and flooding.

The results indicate that the 2-high arrangement only leads to an increase in storage capacity of circa 7% compared to the 1-high arrangement. As storage in a stacked configuration introduces additional cost, hazards and handling complexity, the 1-high arrangement may become the preferred Concept Design option.

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Track 7

A PARAMETRIC STUDY OF THE EFFECT OF REACTOR OPERATING CONDITIONS ON GADOLINIUM PEAK REACTIVITY DETERMINATION FOR BWR UO_2 USED FUEL TRANSPORT AND STORAGE

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Transport and storage (dual purpose) casks for BWR UO_2 used fuel assemblies are usually designed, regarding criticality safety analysis, with the assumption of fresh fuel (furthermore an isotopic composition corresponding to the most penalizing fresh fuel enrichment is considered) and neglecting the presence of neutron integral burnable absorbers, such as gadolinium, in some fuel rods.

During the last decade, Orano TN decided to investigate new methods, due to the continuous increase of the enrichment of modern BWR UO_2 fuel assemblies. The aim of this approach is to limit both the increase of the neutron poison content in new basket designs and to enhance the performance of the casks with existing baskets. Orano TN's strategy consists in taking credit for the presence of gadolinium in the BWR UO_2 fuel assemblies. However, this approach requires defining a methodology that ensures the conservatism of both used fuel isotopic compositions and criticality calculations. Indeed, depletion calculations, and hence criticality calculations for the purpose transportation and storage cask of BWR UO_2 fuel assemblies, depend on several local core conditions. These are the coolant void fraction, the presence of control blades during operation and some relevant irradiation parameters such as the specific power, the coolant temperature, the moderator temperature and so on. As well, the impact on the cask reactivity of the environment in the core of the depleted fuel assembly is analyzed in this parametric study.

This paper presents the results of sensitivity calculations performed to analyze the influence of main core parameters on BWR UO_2 used fuel transport and storage cask reactivity.

DEFINING SAFE FISSILE MASS LIMITS FOR TRANSPORT PACKAGES CARRYING INTERMEDIATE LEVEL WASTE

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Radioactive Waste Management Ltd (RWM) is responsible for developing a geological disposal facility (GDF) for the UK's higher activity waste. In support of this RWM has developed and maintains a generic Transport Safety Case (gTSC) to demonstrate that radioactive waste packaged at present will be safe to transport to the GDF in the future. There are a number of Intermediate Level Waste (ILW) streams planned for disposal at a GDF which have been identified as requiring transportation under the IAEA Transport Regulations as a Type B(U)F or B(M)F transport package due to their fissile content. Accordingly RWM have defined Safe Fissile Masses (SFMs) for a range of transport package designs.

Initially diversity in the composition of these wastestreams was allowed for by grouping wastes into bands and defining a SFM for each band. For example; bands covering a range of isotopic composition such as irradiated natural/slightly enriched uranium, low enriched uranium, etc. The uncertainties in the disposition of material within a package were taken into account by pessimistically assuming the material forms an optimal configuration. However this led to the definition of very restrictive SFM.

To reduce these pessimisms RWM working together with International Nuclear Services (INS) and Sellafield Ltd (SL) has developed an approach to group wastes by material distribution. Material distribution bands have been defined for a worst-case of unlimited non-uniformity (corresponding to the historical limits for an optimal configuration) and for material distributions that allow credit to be taken for a known degree of mixing. The approach is supported by compliance rules to enable wastes to be assigned to an appropriate distribution band. However, this approach resulted in number of parameters that were sub-optimal to the extent that would normally be acceptable to the IAEA Transport Regulations, i.e. full optimisation of all parameters.

Optimising every parameter would require considerable amounts of modelling and interpretation at large cost resulting in minimal gains in safety for some parameters. RWM engaged with the UK Competent Authority (the Office for Nuclear Regulation (ONR)) to resolve the issue. Following discussions ONR escalated the problem to the IAEA who have now added additional guidance to accompany the IAEA Transport Regulations. The guidance now states "Where the number of possible parameters is very large the probability of them all achieving their most reactive value during normal or accident conditions of transport may be vanishingly small. In such cases it may not be necessary for a criticality safety assessment to assess all possible permutations provided the Competent Authority is satisfied that criticality safety has been adequately demonstrated." This paper describes an approach to interpret the new guidance and deliver the benefit of defining less restrictive SFMs.

EVALUATION OF CRITICALITY SAFETY MEASURES FOR FUEL STORAGE OF CRITICAL ASSEMBLIES IN STACY

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For compliance with the new regulatory requirements in Japan, the Static Critical Experiment Facility (STACY) has been remodeling the existing fuel storages. In the remodeling, the existing fuel storage spaces, to which shape and dimension management are applied, are designed to add a neutron absorber for the critical control, taking into account the case of shape and dimension collapse. In order to confirm the validity of the criticality safety design, subcritical calculations were performed. In the calculations, the Japanese Evaluated Nuclear Data Library, JENDL-3.2, was used to cross reference the data. The neutron multiplication factor was calculated using a continuous-energy Monte Carlo code, MVP, and PIJ code in the SRAC code system. It has been confirmed from the results that all fuel storages comply with the safety criteria required to ensure subcriticality.

Track 9

STOCHASTIC BEHAVIOUR OF A CRITICALITY EXCURSION WITH LOW SOURCE

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We consider the case of a prompt critical excursion in a fissile solution with a linear reactivity profile without feedback and a low source. The problem is to evaluate the reactivity level when the number of fissions corresponding to the boiling point of the solution is reached. This number is usually obtained by solving the kinetic equations which represent the average behavior for the time evolution of the neutron and precursor populations. However when the neutron source is weak and the populations are small, the fluctuations in time and from one realization to another are no more negligible. In this case a probabilistic description is necessary. Stochastic simulations can be performed using Monte Carlo analogue methods or by deriving and solving stochastic kinetic equations (Kolmogorov backward master equations). We consider the application of both methods to a given point model test problem.

REVIEW OF CRITICALITY ACCIDENT ALARM SYSTEM REQUIREMENTS IN GEOLOGICAL DISPOSAL

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The United Kingdom Geological Disposal Facility (UK GDF) will receive, transfer and emplace containerised nuclear waste in an underground storage facility. As part of these operations it is necessary to consider the need to install a Criticality Accident Alarm System (CAAS) or a Criticality Incident Detection System (CIDS) for this operational phase.

Work has been undertaken to study international approaches for similar geological disposal projects and surface and near-surface waste storage facilities. These studies were performed to provide insight and precedent to better inform and guide the future assessment of CAAS/CIDS for the UKGDF. This paper summarises the considerations and findings made during these studies. The main conclusion from the studies is that, on the evidence assembled, a CAAS/CIDS omission case is likely to be supportable for a UK GDF. The majority of examples studied successfully presented cases for CAAS/CIDS omission. In those cases, the majority were based on arguments of low probability given failure of any or all controls based on human agency, active engineered or operational controls. The UK GDF is currently planned on few operational controls and robust resilience in the nature of waste containment. It therefore lends itself to likely similar approaches of arguments of low probability of a criticality accident.

Track 10

LITTLE CRITICALITY: A HELPFUL TOOL FOR A CRITICALITY SAFETY ENGINEER

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A criticality safety engineer often needs to consult nuclear parameters like infinite multiplication factor or materials buckling and sub-critical limits (diameter, thickness, volume, and mass).

Those simple values help him:

- Comparing various fissile media,
- Pre-designing individual fissile units,
- Specifying safe limits for the design and operation of process facilities,
- Preparing more complex calculations.

It is difficult for the criticality safety engineer to find the adequate reference quickly, to estimate the impact of the modification of a sub-critical limit or to advise mechanical engineer or process's engineer in the criticality safety design of equipment. To help him, Orano Projects chose to design software for criticality safety engineer. A proof-of-concept was performed two years ago and culminated in the design of a Windows-based software called Little Criticality for:

- Accessing a database of nuclear parameters and sub-critical limits,
- Extrapolating those sub-critical limits to other criteria or other geometries (limits for the sphere, infinite cylinder, infinite slab, and infinite annular cylinder),
- Calculate effective multiplication factors for simple geometries (sphere, cylinder, and slab).

The design of this software has been made very ergonomic. Quick access to a database allows us finding the right technical note easily for our criticality safety analysis. The database can be filtered by chemical composition, density, uranium enrichment, plutonium isotopic vector, moderator, calculation geometry, calculation criteria, and calculation reflector. It is possible to plot curves to compare various sub-critical limits and to export these sub-critical limits in a CSV file.

Extrapolating sub-critical limits and calculating effective multiplication factors for simple geometries allow us specifying the proper criticality safety calculation quickly. To extrapolate sub-critical limits, this software uses the database to calculate extrapolation values, linear interpolation, and hand-calculation methods.

The object of the paper is to present Little Criticality, describe the extrapolation method and quantify the error due to the use of this simple method.

NEUTRONICS LIVENED UP BY COMPUTER

PAUL REUSS

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A set of about fifty exercises, illustrations or animations of neutronics has been developed as a complement and an illustration of the more traditional approaches as lectures, practical works and written materials. The use of a computer allows an individual work with interactivity and movements. This is particularly useful for the analysis of the neutron migration, of the criticality risk, and of all the dynamic processes which characterize a nuclear reactor.

CRITICALITY SAFETY TRAINING APPROACH IN A FUEL ASSEMBLY MANUFACTURING SITE: TESTIMONY AND CONSIDERATIONS

JEAN-FRANÇOIS PAPUT

Romans' Framatome plant, France

This paper aims at describing the training mission of a criticality manager (ICC : "Ingénieur Criticien de Centre") in a fuel assembly manufacturing site.

Romans' Framatome site is dedicated to manufacturing enriched uranium fuel assemblies for nuclear power reactors and fuel elements for research reactors.

Generally speaking, the special characteristics of the implemented processes are the following :

- accessibility to nuclear equipment and to fissile materials,
- many hand-operated processes (manufacturing processes, nuclear material transfer...),
- no severe radiological constraints, especially in terms of ambient irradiation.

Concerning the training objectives, the first message for criticality safety is that the procedures validated by the ICC must be strictly followed.

A further objective is to develop a questioning attitude. In case of hazards, questionings, difficulties encountered, abnormal facts, personnel have to raise deviations to their hierarchy.

The training aims at developing the knowledge of the personnel but the purpose is not to urge them to analyze on their own an abnormal situation in order to avoid inappropriate actions.

We need to call upon the ability of people who attend a criticality training to identify potential topics related to the criticality risks and to report information to the hierarchy.

For these aspects, the operations personnel are key in guaranteeing safety. Periodic training is fundamental to remind everyone of the specific criticality concerns with enriched uranium in the case of Romans' facilities and so why people have to strictly adhere to safety rules.

The worst enemy in our activities is the routine that can exist and lead to think that an accident is highly unlikely.

Workshops

Two workshops will be held on Thursday, September 19 afternoon in the Rooms 3 and 4 (Level -3 of the Convention Center).

You can register for these workshops from Sunday, September 15 at the Information Desk (Level -1).

Workshop 1

Second Level Criticality Modelling with CRISTAL Package: Enhancing Criticality Safety Assessments for Industrial Applications

*-3 Room **3** • Duration: **2h30** • Maximum number of participants: **12***

MODERATORS
IRSN, CEA, ORANO and Framatome

OBJECTIVES
Use CRISTAL V2 package to perform criticality calculations and have a hands-on training course of CRISTAL V2 codes.
Design annular tank for U solution storage using CRISTAL V2 package

MATERIALS
Computers with CRISTAL software will be provided by the ICNC 2019 organization.

Workshop 2

Enhancing Validation of Nuclear Criticality Safety Calculations with ICSBEP Handbook and NEA Tools

*-3 Room **4** • Duration: **3h30** • Maximum number of participants: **45***

MODERATORS
John Bess (INL), Ian Hill (NEA), Shuichi Tsuda (NEA)

OBJECTIVES
Provide participants with the opportunity to examine and discuss the ICSBEP benchmark evaluation process, and have a hands-on training course of NEA tools of database DICE, NDAST, and SFCOMPO.

MATERIALS
Participants should provide their laptop computers (with Windows or Mac system)

Technical Tours

Through the technical tours proposed, visitors will be able to visit the French fuel cycle, from the manufacture of PWR assemblies at Framatome Romans-sur-Isère facility to the reprocessing of fuels at ORANO Cycle La Hague facility and finally, the manufacture of MOX fuels at ORANO Cycle MELOX. At last, the CEA visit gives them the opportunity to visit one of the rare experimental reactor under construction, the RJH, and the LECA-STAR laboratory.

framatome

Framatome Romans

The Framatome Romans site is a fuel fabrication plant. Located about 5 km of Romans-sur-Isère (south of France), the Romans-sur-Isère facility is dedicated to the fuel assemblies fabrication for nuclear power reactors and research reactors.

DE LA RECHERCHE À L'INDUSTRIE
cea

CEA Cadarache

Cadarache is one of the nine research centers of CEA (French Alternative Energies and Atomic Energy Commission). Its activities are dedicated to nuclear energy (fission and fusion), new technologies for energy and biology.

orano

ORANO Cycle MELOX

The Melox plant of the Orano group, located at the Marcoule site, fabricates MOX fuel for the reactors of nuclear power plants in various countries. Made with a blend of uranium and plutonium oxides, MOX fuel recycles plutonium recovered from used fuel. With more than 2,700 metric tons of heavy metal produced as of the end of 2017, Orano Melox is the world's leading producer of MOX fuel.

ORANO Cycle La Hague

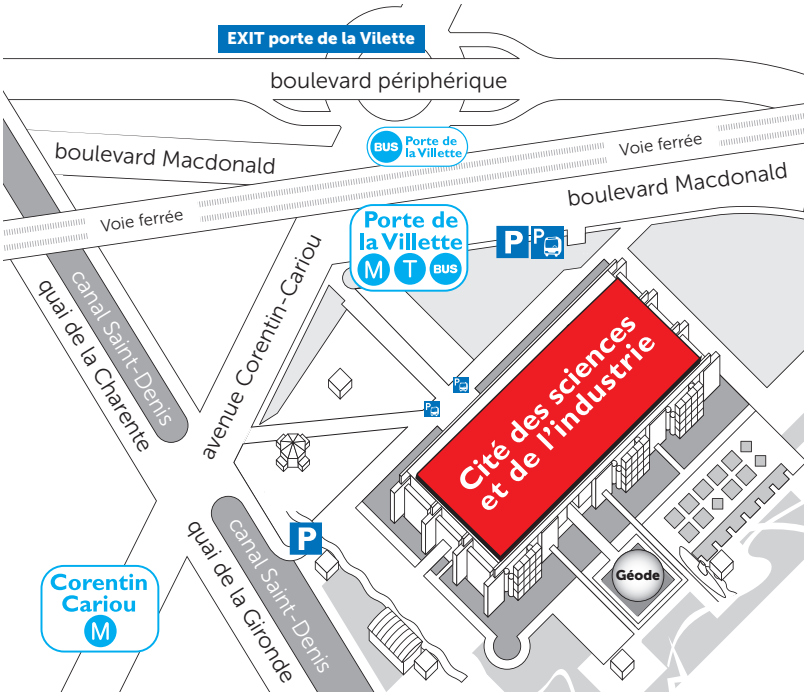
As part of its nuclear activities, Orano La Hague site provides the first steps of used nuclear fuel recycling. This plant is the largest nuclear fuel reprocessing facility in the world. Of about 96 % of used fuel can be recovered providing nuclear reactor reprocessed fuel for electricity production.

As a whole more than 34,000 tons of fuel have been reprocessed at Orano La Hague. Orano nuclear fuel reprocessing activities are at the hand of high qualified employees. Orano employees are committed to sustainable and responsible approaches by optimizing the use of nuclear energy thru nuclear fuel reprocessing.

General Information

ADDRESS OF THE CONVENTION CENTER

Cité des sciences et de l'industrie
30, avenue Corentin-Cariou, 75019 Paris, FRANCE



ACCESS

By Air

- ***From Roissy Charles de Gaulle airport:***
Take RER B, from the stop *Aéroport Charles de Gaulle* to the stop Gare du Nord
Then Metro 5: until the stop *Stalingrad*
Then Metro 7: until the stop *Porte de la Villette*
- ***From Orly airport:***
Take Tram 7: from the stop *Aéroport d'Orly* to the stop *Villejuif – Louis Aragon*
Then Metro 7: until the stop *Porte de la Villette*

By Train

- ***From Montparnasse station:***
Take Metro 4: from the stop *Montparnasse Bienvenue* to the stop *Gare de l'Est*
Then Metro 7: until the stop *Porte de la Villette*
- ***From Gare du Nord station:***
Take Metro 5: from the stop *Gare du Nord* to the stop *Stalingrad*
Then Metro 7: until the stop *Porte de la Villette*
- ***From Gare de Lyon station:***
Take Metro 1: from the stop *Gare de Lyon* to the stop *Bastille*
Then Metro 5: until the stop *Gare de l'Est*
Then Metro 7: until the stop *Porte de la Villette*
- ***From Gare de l'Est station:***
Take Metro 7: until the stop *Porte de la Villette*
- ***From Saint Lazare station:***
Take Metro 3: from the stop *Saint Lazare* to the stop *Opéra*
Then take Metro 7: until *Porte de la Villette*

By Public transport

- ***Metro :***
Line 7, *Porte de la Villette* station
- ***Bus :***
Lines 139, 150, 152, *Porte de la Villette* stop
- ***Tram :***
T3b (Porte de Vincennes – Porte de la Chapelle) *Porte de la Villette* stop

By Car

Take the north Paris ring road (Périphérique) and exit at Porte de la Villette.
Coach park (bus, minibus, etc.) with paid access (10 minute free drop-off), entrance on boulevard Macdonald only.
For more information and booking inquiries, call +33 (0)1 40 05 79 90.

Paying car park: with 1500 spaces, 32 spaces reserved for disabled people, entrances on boulevard Macdonald and Quai de la Charente.
Open everyday, 24 hours a day, direct access.
Max. height: 1.80 metres.

CONVENTION CENTER PRACTICAL INFORMATION

Registration / Information Desk

The front desk, located in Level -1, will be open for registration and information during the following hours:
- Sunday, September 15: 15:00 – 20:00
- Monday, September 16: 8:00-18:00
- Tuesday, September 17: 8:00-18:00
- Wednesday, September 18: 8:00-18:00
- Thursday, September 19: 8:00- 17:00

Badges

Please note that the attendees are required to wear and display their conference badge at all time in the Convention Center. Access to all venues will be checked. Anyone who is not registered to the conference will not be allowed to access to the Convention Center. On site late registrations to the conference are possible at the Registration / Information desk.

Wifi

Free Wifi is available in the Convention Center:
SSID: **11th ICNC 2019**
Password: **irsn@2019**
Once connected, you will be redirected directly on the ICNC 2019 website.

Speakers

A speaker preparation desk will be location in Room 2 (Level -3).
Speaker preparation room operating time is:
- Sunday, September 15: 15:00 - 18:00
- Monday, September 16: 8:00 - 18:00
- Tuesday, September 17: 8:00 - 18:00
- Wednesday, September 18: 8:00 - 18:00
- Thursday, September 19: 8:00 - 12:30

Speakers are required to provide the secretary team with their presentation as soon as possible at the latest during the break preceding the session.

Posters

Printed posters will be displayed in (Level -1) for the entire duration of the conference.
A poster session is planned on Tuesday, September 16, between 16:10 and 17:50. Authors are also encouraged to discuss during coffee breaks.

Coffee Breaks

Complimentary tea, coffee and pastries will be served (Level -2) at the times specified in the program.

Lunches

Lunches will be served at the times specified in the program in the dedicated place (Level -2).

Welcome Reception

The ICNC Welcome Reception will be held on Sunday, September 15, 18:00-20:00 (Level -1).

Gala Dinner

The Gala Dinner will be held on Wednesday, September 18, 19:00-23:00 at the Explora Forum (top floor of the Cité des sciences et de l'industrie).
Please note that the access to the Gala Dinner is restricted to attendees having purchased Full conference pass or One-day registration for Wednesday, September 18, or for accompanying persons who have registered and payed for that.
If you are not registered for the Gala Dinner, please inform us by Wednesday at 12:00 at the latest. Late registration will depend on availability.

Mobile Phone

Attendees are requested to switch their mobile phone on silent mode when entering the sessions.

Language

English is the official conference language.

Social Media

You may tweet about the conference using the hashtag: **#ICNC2019**

Restaurants around

In the Cité des sciences et de l'industrie :
Burger King® (Level -2)
Biosphère (Level -1)
Atmosphère (Ground Floor 0)
Rest'O (Level -2)

Sightseeing Excursions

THE VENUE CITÉ DES SCIENCES ET DE L'INDUSTRIE:

In the heart of the multi-cultural site of La Villette in Paris, the Cité des sciences et de l'industrie has provided a bridge between science, society and technology, since it was created in 1986. A place for meeting and exchange, the Cité des sciences is keen to ensure that the development of science, technology, industrial expertise and that the surrounding issues are accessible to all. In order to achieve this, the Cité des sciences et de l'industrie offers rich and varied cultural provisions, suitable for audiences of all ages.

Around the Cité des sciences et de l'industrie: Parc de la Villette

Parc de la Villette is the largest urban cultural park in the capital, designed by architect Bernard Tschumi. The 55 hectares of the park, 35 of which are outdoors, combine nature and modern architecture, recreational areas and spaces for children and adults, cultural sites and entertainment venues. Open from 6 a.m. to 1 a.m., it can be reached by metro, bus, foot, bicycle and even by boat. You can stroll along the canal de l'Ourcq or enjoy the many green spaces, ponds and fountains. There are many cultural places: Cité des sciences et de l'industrie, Géode, Zénith de Paris, Cité de la Musique, Philharmonie de Paris.

- ***Opening***
  The parc de la Villette is open every day, 6:00 to 1:00.
- ***Getting There***
  Only a few minutes walk from the Cité des sciences et de l'industrie.

DISCOVER PARIS

Eiffel Tower

The Eiffel Tower construction in 2 years, 2 months and 5 days was a real technical and architectural performance. "Utopia realized", a technological feat, it was at the end of the 19th century the demonstration of the French genius embodied by Gustave Eiffel, a highlight of the industrial era. It was an immediate success. The Eiffel Tower has been listed as a historic monument since 24 June 1964 and has been a UNESCO World Heritage Site since 1991. The Eiffel Tower will allow you to admire one of the most beautiful 360° views of Paris.

- ***Opening***
  The Eiffel Tower is open every day, 09:30 to 23:45 (last access at 23:00).
- ***Getting There***
  By public transport take Metro Line 6 and get off at Bir-Hakeim*; or RER Line C and get off at* Champs de Mars-Tour Eiffel.

Arc de Triomphe

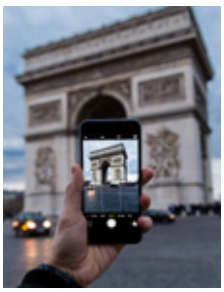
A major venue for major national events. Wished by Napoleon I in 1806, the Arc de Triomphe was inaugurated in 1836 by the King of the French, Louis-Philippe, who dedicated it to the armies of the Revolution and the Empire. The Unknown Soldier was buried on the median in 1921. The flame of remembrance is rekindled every day at 18:30. Beyond the historical aspect, the Arc de Triomphe will offer you, from its panoramic terrace, one of the most beautiful views of Paris, with a breathtaking view of the Champs Elysées.

- ***Opening***
  The Arc de Triomphe is open every day, 10:00 to 23:00 (last access at 22:15).
- ***Getting There***
  By public transport take Metro Line 1, 2 or 6 and get off at Charles-de-Gaulle-Etoile.

The Louvre

The Louvre, the former royal palace and the most visited museum in the world, is located in one of the most prestigious districts of the French capital, in the first arrondissement between rue de Rivoli and quai François Mitterrand. The Louvre is the largest art and antiques museum in the world. From room to room, the former royal palace unveils its masterpieces: the Mona Lisa, the Raft of the Medusa, the Venus de Milo, the Victory of Samothrace...

- ***Opening***
  The Louvre is open every day except Tuesday, 09:00 to 18:00.
- ***Getting There***
  By public transport take Metro Line 1 or 7 and get off at Palais-Royal -Musée du Louvre*; or Metro Line 14 and get off at* Pyramides.



Musée d'Orsay

Discover one of the most beautiful collections of impressionist art in the world with masterpieces by Monet, Renoir, Degas and Cézanne. The Musée d'Orsay houses the French national collection of impressionist, post-impressionist and Art Nouveau works. Its richness and wonders make it one of the most important cultural museums in the world. It is rich in history, both through the works on display and the splendor of the building itself.

- ***Opening***
  The Musée d'Orsay is open every day except Monday, 09:30 to 18:30 (last access at 17:00).
- ***Getting There***
  By public transport take Metro Line 12 and get off at Solférino *or take RER Line C and get off at* Musée d'Orsay.

Centre Pompidou

The Centre Pompidou is a must-see museum in Paris... Located in the heart of Paris in the Marais district, the Centre Pompidou is an icon of 20th century architecture, timeless and resolutely modern.

Unavoidable exhibitions: masterful figures and founding movements of the 20th century art history as well as the greatest artists of the contemporary scene.

The Centre Pompidou offers one of the most beautiful views of the monuments of Paris.

A typically Parisian, multidisciplinary and unique place in the world!

- ***Opening***
  The Centre Pompidou is open every day except Tuesday, 11:00 to 21:00.
- ***Getting There***
  By public transport take Metro Line 1 or 11 and get off at Hôtel de ville*; or Metro Line 4, 7 or 14 and get off at* Châtelet.

Bateaux Mouches

The Compagnie des Bateaux-Mouches® was founded in 1949 by Jean Bruel. At the beginning, a unique steamboat, a souvenir of the 1900 Universal Exhibition. Today a modern fleet of 15 ships that welcome nearly 2.5 million people each year, Parisians and tourists passing through. Get on boat to take a grand tour and admire, along the Seine, for more than an hour, the capital's emblematic monuments. Visiting Paris by boat allows you to change your point of view and discover the historic heart of the capital from a different angle.

- ***Opening***
  A departure approximately every 30 minutes from 10:00 to 22:30. The duration of the cruise is about 1h10.
- ***Getting There***
  By public transport, take Metro Line 9 and get off at Alma-Marceau*, The pier is located near the Pont de l'Alma on the Port de la Conférence.*

DISCOVER THE REGION

Château de Versailles

The Chateau de Versailles, whose origins date back to the 17th century, was successively a hunting lodge, a place where power was exercised and since the 19th century a museum. Composed of the Palace, the gardens, the Park, the Trianon estate and a few outbuildings in town, the Domaine now covers more than 800 hectares. Classified for 30 years as a World Heritage Site, the Chateau de Versailles is one of the most beautiful examples of French art. Discover a place that is representative of French history. From the seat of power to a museum of the history of France.

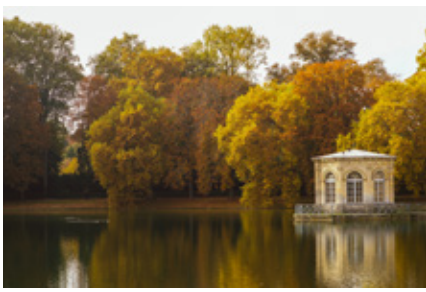
- ***Opening***
  The Château de Versailles is open every day except Monday, 09:30 to 18:30 (last access at 18:00).
- ***Getting There***
  By car from Paris: *Motorway A13, then take exit no.5 Versailles Centre and follow the signs for the Chateau de Versailles. Approximately 28 Km.*
  By public transport: *Take RER Line C arrives and get off at Versailles Château-Rive Gauche train station, just 10 minutes walk to the Palace.*

Château de Fontainebleau

Immerse yourself in over 1,500 rooms and 130 stunning acres of French history and greatness.

At the Chateau Fontainebleau, take a stroll through the grand interiors and opulent gardens of the « House of the centuries, true dwelling of Kings ». A UNESCO World heritage site, this is the only French royal and imperial Chateau to have been continuously inhabited for seven centuries. Here, you'll unearth multiple galleries, chapels, museums and theatres in what is an unparalleled view of French politival, royal, art and architectural history.

- ***Opening***
  The Château de Fontainebleau is open every day except Tuesday, from April to September: 09:30 to 18:00 (last access at 17:15).
- ***Getting There***
  By car from Paris: Motorway A6 (Porte d'Orléans or Porte d'Italie), then take exit Fontainebleau and follow the directions for "château". Approximately 67 Km.



Notes

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Paris, France

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